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ABSTRACT

This report provides an assessment of safety management, safety review, and safety methodology employed by the Department of Energy (DOE) and private contractors. Chapter 1, "The DOE Safety Framework," examines safety objectives for production reactors and processes to implement the objectives. Chapter 2, "Technical Issues," focuses on a variety of potential vulnerabilities to severe accidents including acute aging phenomena, evaluation of potential accidents, power operating limits, confinement systems, and the treatment of radioactively contaminated liquid effluents at the reactors. Chapter 3, "Strengthening the Technical Basis of Reactor Safety Management," identifies ways in which the DOE approach to management of the safety of the reactors can be improved. Each subsection consists of conclusions and recommendations. Over 500 documents are listed in the bibliography. Appendices include a task statement; a list of documents about power operating limits; technical discussions on confinements, aging, effluents, cermet fuel, hydrogen generation during accidents, and the DOE safety system; safety-related provisions; and an introduction of committee members. (YP)

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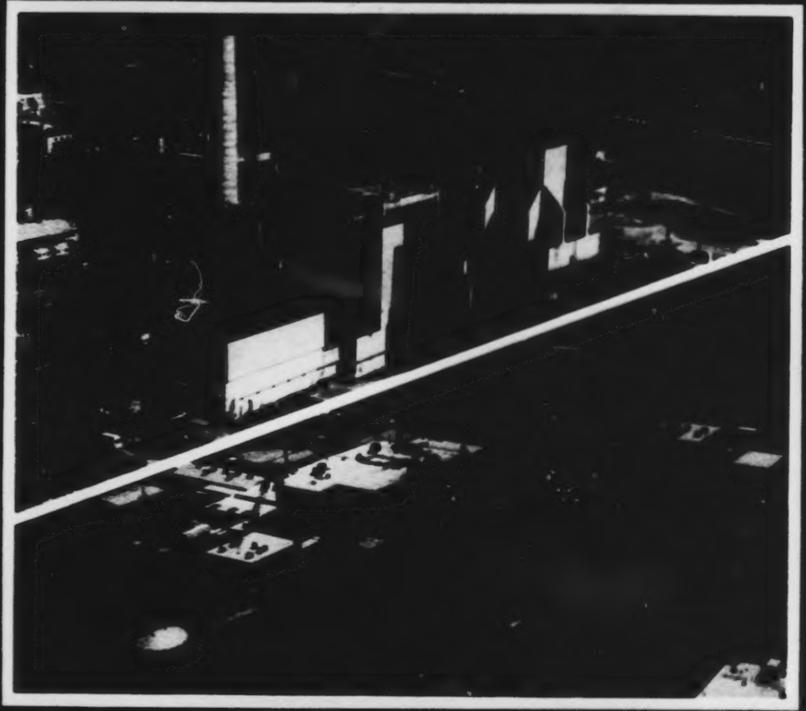

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front cover: *(upper photo) One of the three Savannah River reactors; (lower photo) the N Reactor.*

back cover: *Computer graphics representation of the projected growth of the high-flux zone of the N Reactor's graphite moderator under two different operating regimes.*

SAFETY ISSUES AT THE DEFENSE PRODUCTION REACTORS

A Report to the U.S. Department of Energy

Committee to Assess Safety and Technical Issues at DOE Reactors
Commission on Physical Sciences, Mathematics, and Resources
and
Commission on Engineering and Technical Systems
National Research Council

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Preface

Shortly after the April 1986 nuclear accident at the Chernobyl Nuclear Power Station in the Soviet Union, Secretary of Energy John S. Herrington requested that the National Academy of Sciences and the National Academy of Engineering provide an independent assessment of the implications of the accident for the safe operation of 11 of the Department of Energy's (DOE) larger reactors. In response, the Academies formed the Committee to Assess Safety and Technical Issues at DOE Reactors, which in August 1986 began the study requested by the Secretary.

This report fulfills a part of the Secretary's request. It provides an assessment of safety and technical issues at four of the eleven Class A reactors: the K, L, and P reactors located at the Savannah River Plant in South Carolina and the N Reactor located on the Hanford Nuclear Reservation in Washington. (The C Reactor, also located at the Savannah River Plant, was retired in January 1987.) These reactors are referred to as "defense production" reactors because they are operated primarily to supply the Department of Defense with plutonium and tritium for use in nuclear weapons. A subsequent report will examine the remainder of the DOE's large reactors.

The N Reactor has operated for nearly 25 years, and the reactors at the Savannah River Plant for over 30 years. During this time, none of the reactors at either site has had an accident approaching the severity of those at either Three Mile Island or Chernobyl. Nevertheless, the scale and consequences of the Chernobyl accident should serve as the impetus for a reexamination of all reactors. The review reported in this document was conducted by a committee whose members are experienced in reactor safety, particularly in the commercial and naval reactor fields.

In conducting the study, the committee reviewed extensive documentation from the DOE and its contractors, including departmental orders, testimony before Congress, safety analyses and incident reports, correspondence, audit and surveillance reports, minutes of meetings, and other documents—all extending back several years. The committee also made several site visits to Savannah River and to the N Reactor in order to interview individuals engaged in management and operations. The committee held a number of meetings at which it heard the views of several members of a panel of experts (generally known as the Roddis panel) that conducted an earlier review of the N Reactor for the DOE. The committee also conducted a series of public hearings to hear the views of citizens in the immediate vicinity of the plants. The committee is grateful to all those who participated.

One of the committee members, Kai Lee of the University of Washington, is also a member of the Northwest Power Planning Council. This council is responsible for formulating forecasts and plans for the Pacific Northwest region. In view of the appearance of a possible conflict of interest between his role on this committee and his responsibilities on the council, Professor Lee has chosen to take no part in any of the committee's deliberations on the reactors at Hanford.

Several events occurred in the course of the study that have had an impact on the overall status of the production reactors. In late 1986, Congress failed to authorize funds requested by DOE for the purpose of extending the life of the N Reactor to the year 2000. In November 1986, the Department of Energy lowered the operating power of the Savannah River reactors in response to uncertainties regarding the capability of the reactors' emergency core cooling systems. On January 6, 1987, the N Reactor entered into a prolonged outage in order to implement changes in administrative procedures, to conduct tests, and to install additional safety equipment. The shutdown was for the express purpose of responding to the many recommendations that resulted from various post-Chernobyl reviews of the reactor. In the spring, Congress considered legislation that would prevent the N Reactor from ever operating again unless assurances were forthcoming as to the plant's safety. Late in the spring, the Secretary of Energy indicated that the Department intended to submit a request for funds in the FY 1989 budget for the design of new production reactors, and around the same time the Department made a request that the National Research Council separately study the viability of selected alternative technologies for such a reactor.

These changes complicated the committee's task. Moreover, in response to particular issues associated with events at Savannah River and Hanford, the committee prepared two letter reports that were submitted to the Secretary of Energy; these are in Appendixes C and G, respectively, to this report.

In language adopted in the House version of the FY 1988 defense authorization bill, Congress requested that the Academies produce a report relating to the safety of operation of the N Reactor. The committee believes this report is responsive to that request. Also late in the spring, members of Congress discussed with officials of the National Research Council the possibility of the Research Council evaluating whether the N Reactor was "safe" to restart. This report does *not* reach any conclusions with respect to restart. Although the committee brings to its task a wealth of experience in the field of nuclear safety, it has neither the legal authority nor the capacity to conduct the in-depth scrutiny that would be necessary to judge the overall safety of any of DOE's reactors. This report should be considered an examination by a number of outside experts of a set of particular technical issues and uncertainties, and an evaluation of the *conceptual* soundness of DOE's approach to reactor safety. Both the strengths and weaknesses highlighted in the report should be understood in that light.

The committee has sought to recommend improvements in production reactor safety in a variety of technical areas. But the report does not cover all technical issues important to safe reactor operation. In particular, the committee does not address such issues as safeguards and security, fuel handling and transportation, occupational safety, fire protection, quality assurance, or waste management.

In its review of safety issues at the production reactors, the committee identified management as a major area of concern. A discussion of the committee's views is presented in Chapter 3. The discussion in Chapter 3, however, is not intended to address how DOE should manage the safety and production challenges of the defense production enterprise as a whole. The production of nuclear weapons involves facilities at half a dozen sites containing reactors, chemical separation plants, waste management facilities, transportation networks, and other activities. DOE's management responsibility is to operate this complex of activities safely while meeting national requirements for the products. The assurance of the safety of the production reactors is but one element of DOE's responsibility.

Contents

EXECUTIVE SUMMARY xiii

INTRODUCTION 1

1 THE DOE SAFETY FRAMEWORK 10
  The Safety Objective, 11
  Department Orders, 14
  Verification of Compliance, 18
  Technological Vigilance, 19

2 TECHNICAL ISSUES 22
  Acute Aging Phenomena, 23
  Maintenance and Plant Modernization, 28
  Power Operating Limits, 33
  Probabilistic Risk Assessment, 38
  Severe Accident Evaluation, 40
  Confinement Systems, 49
  Hydrogen Generation and Mitigation, 52
  Cermet Fuel, 56
  Human Performance, 58
  Liquid Effluents, 64
  Emergency Planning, 66

3 STRENGTHENING THE TECHNICAL BASIS OF REACTOR SAFETY MANAGEMENT 70
The Role of the Department of Energy in Historical Context, 70
The Current Management System, 73
Principal Areas for Improvement, 75

BIBLIOGRAPHY 84

APPENDIXES

A. Statement of Task, 119
B. Confinements: Technical Discussion, 121
C. Documents on Power Operating Limits at Savannah River Plant, 134
D. Aging: Technical Discussion, 165
E. Effluents: Technical Discussion, 177
F. Cermet Fuel for SRP Reactors: Technical Discussion, 186
G. Hydrogen Generation During Accidents at N Reactor: Technical Discussion, 192
H. Structure of the DOE Safety System: Technical Discussion, 222
I. Selected Reactor Safety-Related Provisions of the Du Pont-DOE and UNC-DOE Contracts, 247
J. Biographical Sketches of Committee Members, 256

Executive Summary

The United States produces plutonium and tritium for use in nuclear weapons at the defense production reactors—the N Reactor in Washington and the Savannah River reactors in South Carolina. This report reaches general conclusions about the management of those reactors and highlights a number of safety and technical issues that should be resolved. The report provides an assessment of the safety management, safety review, and safety methodology employed by the Department of Energy and the private contractors who operate the reactors for the federal government.

The report is necessarily based on a limited review of the defense production reactors. It does not address whether any of the reactors is "safe," because such an analysis would involve a determination of acceptable risk—a matter of obvious importance, but one that was beyond the purview of the committee. It also does not address whether the safety of the production reactors is comparable to that of commercial nuclear power stations, because even this narrower question extended beyond the charge to the committee and would have involved detailed analyses that the committee could not undertake. (In addition, the committee did not address such specific issues as restart of the N Reactor or the particular steps necessary to restore full power operation to the

Savannah River reactors.) Nonetheless, the committee hopes that its study will prompt the Department of Energy, others in the Executive Branch, and the Congress to address and resolve the safety and policy issues discussed in the report.

The report is organized in three chapters. Chapter 1, "The DOE Safety Framework," examines the safety objective established by the Department of Energy for the production reactors and the process the Department and its contractors use to implement the objective. Chapter 2, "Technical Issues," focuses on a variety of uncertainties concerning the production reactors, particularly those related to potential vulnerabilities to severe accidents. While reading Chapter 2, it is important to bear in mind that over the years the contractors have emphasized the prevention of accidents, and the production reactors have been operated for more than 25 years without experiencing a major one. Chapter 3, "Strengthening the Technical Basis of Reactor Safety Management," identifies ways in which the DOE approach to management of the safety of the production reactors can be improved.

The report raises many safety questions and considers ways of addressing them. This summary presents the principal conclusions and recommendations of the report as drawn from Chapters 1, 2, and 3.

THE DOE SAFETY FRAMEWORK

The stated ***safety objective*** for the defense production reactors is the achievement of a level of safety comparable to commercial nuclear power plants. However, the committee found a high degree of confusion both within DOE and among the contractor staff concerning the safety objective. The committee concludes that the Department has not clearly articulated, documented, or implemented any specific safety objective for its reactors. While the committee does not criticize the objective of comparability per se, the Department's approach is so imprecise that it fails to provide a clear benchmark against which to determine whether comparability is being achieved. The absence of clear DOE guidance leads the committee to recommend that the Department clarify its safety objective and, in the process, provide a safety objective that is operationally meaningful to DOE and contractor staff as well as understandable to the public. The Department's safety objective should provide a clear foundation on which the implementation of safety can be built.

In support of the safety objective, the Department sets *requirements* for the production reactors through the issuance of Department Orders and through its contracts with the corporations that operate the reactors. However, the committee finds that the Department has failed to specify its safety requirements clearly, has failed to apply them uniformly at the two production reactor sites, and has failed to implement them in a timely manner. The committee recommends that the Department revise its Orders so that they specify requirements clearly and establish deadlines and that DOE and Congress budget funds for implementation.

Following the Chernobyl accident, the Department, on an ad hoc basis, increased its audits of contractor performance. This effort—its recent origin aside—has been commendable. The committee urges the Department to continue to conduct comprehensive audits of contractor performance on a frequent and continuing basis and to require prompt responses to these audits from the contractors in order to help assure reactor safety.

The age and unique designs of the production reactors demand high levels of technical capability in operating, engineering, and support staff. Indeed, increased levels of technological vigilance are needed at the defense production reactors. The committee recommends that the Department and its contractors develop an expanded in-house capability to address, evaluate, and achieve an in-depth understanding of the technical issues associated with the safety of the production reactors. Methods and analytical techniques that are at least comparable in technical sophistication to those used within the commercial nuclear industry should be developed and used. The analyses and supporting tests arising from this effort should be comprehensive in scope and should be subjected to thorough outside peer review.

TECHNICAL ISSUES

The main body of the report is contained in Chapter 2 and consists of evaluations of eleven technical issues pertaining to one or both of the production reactor sites. Key conclusions and recommendations in Chapter 2 relate to the committee's assessment of such issues as acute aging phenomena; evaluation of potential severe accidents; power operating limits; confinement systems; and the treatment of radioactively contaminated liquid effluents at the production reactors. Almost all the issues highlighted in Chapter 2 deal wholly or in part with phenomena that could accompany potential severe accidents at the production reactors. This emphasis derives primarily from the Secretary of Energy's request that the committee examine the Department's reactors "in light of the Chernobyl accident." The committee stresses in the report, however, that attention to the causes and consequences of severe accidents cannot in and of itself assure the safety of the production reactors and must not be allowed to detract from programs essential to managing, operating, and maintaining the production reactors so that severe accidents do not occur in the first place.

One of the key issues identified in Chapter 2 has to do with the age of the production reactors. Several other technical issues discussed in Chapter 2, such as the conduct of severe accident evaluation and probabilistic risk assessments, cannot properly be resolved without first answering the question of how long these aged reactors are to be operated. The level of effort that should be expended to evaluate potential severe accidents or to conduct probabilistic risk assessments must be determined from a realistic appraisal of the remaining service life of the production reactors. Acquiring a detailed understanding of the many potential severe accident phenomena to which these reactors may be subject would be a time-consuming and expensive undertaking; it would make little sense to begin an effort that would not be completed until after the reactors are retired.

The policy and planning process of the federal government has not realistically addressed the aging of the defense production reactors. Among previous determinations that have proven to be misguided are that the N Reactor would soon be retired and therefore need not be modernized and, later, that the life of the N Reactor would be extended through the year 2000, and, in the case of Savannah River, that the reactors could be operated essentially indefinitely.

The discussion of acute aging phenomena in Chapter 2 describes the extent to which the current production reactors display symptoms of acute aging that could affect safety and are likely to limit their useful lives. The level of uncertainty about how long the production reactors can be safely operated is high, and it is unknown whether a new production reactor or alternative means of supplying strategic nuclear materials will be available before each of the existing reactors encounters some age-related problem that precludes safe operation. The committee recommends that if the United States finds it necessary to have a reliable and safe capability for the production of plutonium and tritium for nuclear weapons, then planning for new production reactors or other alternatives should be accelerated.

Chapter 2 also discusses the analysis of loss-of-coolant accidents at the production reactors and the determination of the limits of safe power operation. Based on preliminary analysis of the significant power derating that was put into effect at Savannah River during the course of the committee's work, the Savannah River contractor and DOE believe there is a reasonable expectation that the emergency core cooling systems will prevent core damage in the event of a hypothetical, design-basis loss-of-coolant accident. They are continuing to analyze the problem. Because adequate resources have only recently been devoted to developing a thoroughly documented understanding of the behavior of the Savannah River reactors in a loss-of-coolant accident, the potential for and consequences of early core heatup caused by flow instability in the first few seconds of such an accident at higher power levels remain to be explored. The committee recommends that before restoring full power operation at Savannah River, the Department satisfy itself, on the basis of a rigorous external review, that it has a thorough understanding of the behavior of the reactors in a major loss-of-coolant accident.

The committee also concludes that a thorough understanding of the behavior of the N Reactor in a major loss-of-coolant accident does not currently exist, and recommends that such an understanding be urgently developed. The understanding should be based on state-of-the-art analytical tools, and the tools should have the ability to analyze potential thermal shock and water hammer phenomena. The use of these tools should be accompanied by rigorous quality assurance of the computer codes, as well as experimental validation of the results, particularly with respect to the distinctive horizontal geometry of the N-Reactor core.

On the more general issue of severe accidents and their evaluation, the committee concludes that the existing level of understanding of severe accident behavior for the production reactors is inadequate to permit a realistic assessment of the effectiveness of these designs in mitigating the consequences of severe accidents. The committee therefore recommends that near-term decisions on changes in the design and operation of the production reactors be guided by simple, physically based models and substantial safety margins. For those reactors that have a significant probability of being operated beyond the next few years, the Department should now commence an in-depth program of severe accident model development and validation. Data from the program should be applied in a continuing reexamination of the risk of severe accidents and a review of the design and operation of the plants.

The production reactors differ from commercial nuclear power plants in their use of confinement systems (i.e., systems that aim to control the release of effluents during accidents) rather than reactor containments (systems that aim to totally contain any accident release) as barriers to the release of radionuclides. The committee concludes that, in theory, the confinement approach can provide an acceptable means for mitigating accidents.

The committee also concludes, however, that there are significant uncertainties in the ability of the existing production reactor confinements to mitigate the effects of radionuclide releases that would be expected to occur during severe accidents. This conclusion rests on a variety of factors. For one, the design of the filtration systems that are crucial to mitigating the release of radionuclides during an accident has been based on calculations of the release of radionuclides from degrading reactor fuel. But research performed in connection with commercial nuclear power plants, as well as recent studies of radionuclide release from the Savannah River reactor fuel, show that this "source term" may be greatly underestimated. Moreover, analyses of the loading of the production reactor confinement system filters during hypothetical accidents have not included adequate consideration of nonradioactive particulates. The committee recommends that the Department demonstrate whether the confinement systems at the production reactors have the capabilities (1) to withstand realistic accident loads, and (2) to provide acceptable attenuation of realistic radionuclide releases in the event of a severe accident.

Hydrogen generation and mitigation is also discussed in Chapter 2. Earlier this year, the committee issued a letter report to the Department of Energy concerning the Department's proposed approach to hydrogen mitigation at the N Reactor. (The letter is summarized in Chapter 2 and included in Appendix G.) In the report, the committee reiterates its earlier recommendations that its concerns be addressed during the development of a detailed design to implement the proposed approach at N Reactor, and that the detailed design be subjected to an independent review prior to adoption.

The committee also has concerns about the consequences of hydrogen generation in an accident at Savannah River. The Savannah River contractor has developed and proposed to DOE a concept for an "improved confinement system" that the committee concludes may cause a buildup of hydrogen in a severe accident. The committee suggests that the proposed system be reviewed to evaluate its potential benefits and also its added risks, particularly with respect to the enhanced possibility of hydrogen-combustion events.

During normal operation of the N Reactor, radioactively contaminated liquid effluents are discharged from the confinement into open, unlined basins. These effluents percolate into the earth. Such direct discharges of liquid effluents into the environment would also occur following accidents at either N Reactor or Savannah River. The direct discharge of radioactively contaminated liquid effluents, whether during normal operation or during accidents, poses a safety hazard and represents an environmentally unsound practice. It also represents a violation of the basic confinement philosophy with which the reactors were designed. The committee recommends that the Department establish means to protect the environment from the radionuclide-contaminated liquid effluents discharged during normal operation of the N Reactor. Means should also be found at both Hanford and Savannah River to prevent the direct discharge of contaminated liquid effluents to the environment following an accident.

STRENGTHENING THE TECHNICAL BASIS FOR REACTOR SAFETY MANAGEMENT

In Chapter 3 the committee examines the Department's relationship with its contractors, its organizational arrangements for ensuring the safety of the reactors, and the need for independent oversight. The committee concludes that the Department, both at headquarters and in its field organizations, has relied almost entirely on its contractors to identify safety concerns and to recommend appropriate actions, in part because the imbalance in technical capabilities and experience between the contractors and DOE staff is of sufficient magnitude to preclude DOE from properly performing its audit function. There is a strong need for comprehensive DOE involvement in the operation of the production reactors. The committee recommends that the Department acquire and properly assign the resources and talent necessary to ensure that safe operation is being attained. Changes in DOE staffing levels and budgets may be required to achieve such a capability.

The committee found that the Department's approach to management falls short of reasonable expectation in attempting to cope with the mix of production and safety responsibilities. On occasion, the contractors have pressed for safety upgrades that DOE has then rejected for budgetary reasons. The committee believes that the Department needs to develop a better approach to making budgetary decisions in the face of important safety needs. The committee also recommends that, in strengthening the monitoring of safety at the DOE production reactors, the Department should expand its capability to sponsor research, conduct and review safety analyses, evaluate operations, analyze trends, and assess proposed plant upgrades. The Office of the Assistant Secretary for Environment, Safety, and Health should have a permanent and significant onsite presence with a formal reporting relationship between onsite personnel and headquarters staff. The Office should have more direct access to and involvement in the resolution of key safety issues on a timely and effective basis, as well as a central role in internal budgetary decisions of DOE.

The committee concludes that the Department's safety oversight of the production reactors is ingrown and largely outside the scrutiny of the public. Weaknesses in management of the defense production reactors have led to a loose-knit system of largely self-regulated contractors operating within budgetary constraints imposed by and on the Department.

In light of the conflicting responsibilities of the Department to meet production requirements and assure safety, the committee considered whether to recommend the creation of an entirely new management structure responsible solely for assuring the safety of the production reactors. In lieu of transferring the Department's reactor safety responsibilities to the Nuclear Regulatory Commission or to some new entity also totally outside DOE, the committee recommends that an independent, external safety oversight committee, advisory to the Secretary of Energy, be established. The oversight committee should possess the following features: members should be of recognized stature with expertise covering the full range of disciplines relevant to reactor safety; members should include individuals from outside the DOE community; the committee should have authority to set its own agenda; the committee should be authorized to review both the product and the process of the Department and contractor efforts, including review of design, safety analysis, operations, management, inspection, and enforcement; the committee should be supported by a full-time, technically qualified staff because it could not function effectively without one, and by a budget adequate for obtaining external technical assistance as required; and the bulk of the committee's work should be unclassified and available to the public. The committee further recommends that the Department encourage each contractor to establish a permanent visiting committee of outside experts to review and assess the implementation of safety initiatives and to report to contractor management at the site and at corporate headquarters. In sum, the committee concludes that DOE can accomplish the reactor safety functions assigned to it by Congress if the Department dedicates itself to the task, although the passage of time may demonstrate that more radical measures should be adopted.

Finally, the committee notes that the technological vigilance required to assure safety at the DOE reactors cannot be generated from organizational structure alone. Within even a large, properly structured organization, safety is a reflection of institutional commitment and capability. Leadership at the policy-making level is essential, and dedication to safety must permeate the Department of Energy. DOE has made recent strides in this direction, but its efforts need to be bolstered, institutionalized, and sustained.

Introduction

On April 26, 1986, Unit Four of the Chernobyl Nuclear Power Station in the Soviet Union was destroyed in a nuclear accident much more severe than any previous nuclear accident in history.

In the United States the response was prompt and vigorous, and is continuing. The Secretary of Energy took several immediate steps to review the near- and long-term safety of reactors owned and operated by the U.S. Department of Energy (DOE). At the Secretary's request, the National Academy of Sciences and the National Academy of Engineering formed a committee to assess safety and technical issues associated with the continued operation of the DOE's Class A reactors—those capable of operating at over 20 MW of thermal power (MWT). This report is a product of the committee's efforts. It focuses on the four remaining "defense production" reactors operated by the Department for the purpose of producing nuclear weapons material.

THE DEFENSE PRODUCTION REACTORS

N Reactor

N Reactor is situated on the Columbia River, within the Hanford Nuclear Reservation, in the south central part of the state of Washington. It is a pressurized, light-water cooled, graphite-moderated reactor rated at 4000 MWT. Steam can be supplied to turbine generators owned by the Washington Public Power Supply System (WPPSS) to produce about 860 MW of electricity. Construction of the N Reactor was authorized in 1958, and the reactor began operation in 1963; it achieved its full power rating in 1964.

The reactor uses metallic uranium fuel clad with zircaloy to produce weapons-grade plutonium. The fuel is held in horizontal tubes and cooled by pressurized water. The zircaloy pressure tubes containing the fuel rods pass through a large graphite moderator stack.

The reactor safety systems include 84 horizontal boron-carbide control rods that can be automatically inserted into the core in one second; a backup system of boron-carbide balls (neutron absorbers) that can be released from hoppers above the core and that fall into the core as a result of gravity; an emergency core cooling system; a graphite and shield cooling system; and a confinement system designed to vent pressures greater than 5 psig that are expected early in any accident and then to filter any subsequent releases of radioactive materials. The reactor is currently operated by Westinghouse Hanford Company.

In recent safety reviews of the N Reactor, attention has been focused on several key issues, including swelling of the graphite moderator stack under neutron irradiation; embrittlement of the pressure tubes under prolonged neutron irradiation and hydrogen exposure; lack of means to detect and mitigate problems that could result from concentrations of combustible gases that might be generated during an accident; performance of the confinement system during an accident; and contamination of the environment by liquid effluents released both routinely and in the aftermath of an accident. Prior reviews of the N Reactor have also prompted DOE to accelerate the performance of a probabilistic risk assessment, a method that can be useful for analyzing some of the foregoing concerns (see Chapter 2); that assessment is still in progress.

Savannah River Reactors

There are three Savannah River production reactors—designated K, L, and P—located on the Savannah River Plant site in South Carolina, near the border with Georgia. The K and P reactors are used mainly to produce tritium, while the L Reactor is currently only used to produce plutonium-239. A fourth reactor, the C Reactor, was retired in January 1987, and a fifth reactor, the R Reactor, was shut down in 1964. L Reactor was restarted in 1985 after a 17-year hiatus. All of the Savannah River reactors are low-pressure, heavy-water cooled and moderated reactors of generally similar design—each with a maximum achieved power of about 2800 MWT. The fuel in the Savannah River reactors is a uranium-aluminum alloy clad in aluminum. Unlike the N Reactor, the plants are not also used for power production. Operation of the first Savannah River reactor began in 1953, at a design power level of about 300 MWT; the plants have been upgraded to the current power level.

The changes in plant design that were necessary to achieve higher power levels led to concerns on the part of the Advisory Committee on Reactor Safeguards, especially with regard to the emergency core cooling systems and the confinements. These changes were reviewed at first on the basis of the military necessity of the plants. Later, the changes were modified according to the need to adhere as closely as possible to the principle of comparability of the safety of the production reactors with the first commercial reactors then undergoing licensing.

The principal reactor safety systems of the Savannah River reactors consist of 427 control rods that can be inserted in less than two minutes; 66 additional safety rods driven by gravity that can be fully inserted into the core in less than two seconds; a backup neutron-absorbing liquid injection system that is designed to be effective in 0.7 seconds; an emergency core cooling system with three independent feed lines (a fourth addition line is being installed in each reactor); and a confinement system with filters to reduce radioactive release in the event of a severe accident. E.I. du Pont de Nemours & Company has operated the Savannah River Plant since the start of construction in 1950.

A number of safety issues has been identified at Savannah River in recent DOE reviews. For example, although the Savannah River reactors have undergone extensive upgrading and modification, they show definite and understandable signs of aging. One of the most serious aging phenomena affects the reactor coolant system. Stress and corrosion have led to the formation of cracks and pinholes at the location of weld-induced stresses. Cracks in the reactor vessel caused C Reactor to be removed from operation in 1985 and retired in 1986.

Uncertainties regarding the efficacy of emergency core cooling during a hypothetical, worst case, loss-of-coolant accident, including those identified by this committee, led to a 50-percent reduction in permissible operating power level at the K, L, and P reactors between November 1986 and March 1987. The ability of the Savannah River confinement system to withstand pressures, temperatures, and aerosol releases associated with a severe accident has also been called into question. Upgrades are under development. Probabilistic risk assessments that evaluate more severe accidents than those contemplated in the design of the reactors are also under way.

COMPARISON OF THE N REACTOR TO THE RUSSIAN RBMK REACTOR

Press reports after the Chernobyl accident called attention to a superficial similarity of the N Reactor to the Russian RBMK reactor (of which the Chernobyl plant is an example): both use graphite as a neutron moderator, both use light water as a coolant, and both were alleged to have similar confinement systems, which allow a filtered release to the environment following an accident, as opposed to the containment structures used by commercial nuclear reactors, which are intended to limit releases to the leak rate of the building. Many of these reports suggested that the two reactors were susceptible to similar accidents. However, these suggestions were misleading because they overlooked important differences between the two reactor designs. The following discussion shows why the committee chose to approach N-Reactor safety in terms of issues identified in the course of the study rather than in terms of any real or apparent parallels with the Chernobyl reactor.

The physical layout and design of fuel and graphite in the RBMK differ fundamentally from the layout and design of fuel and graphite in the N Reactor. The RBMK is a boiling water reactor (that is, under normal operating conditions steam is formed within the fuel tubes), while the N Reactor is a pressurized water reactor in which pressure is maintained at a level that precludes boiling of the coolant within the reactor core under normal operating conditions. The configuration of the graphite and fuel within the RBMK leads to a so-called "positive void coefficient." That is, if power increases for some reason, the reactor heats up and a greater volume of steam is formed within the fuel tubes. This increased steam volume, or "void" in the cooling water, causes power to increase further—a potentially unstable condition. By contrast, the configuration of the fuel and graphite in the N Reactor has been designed deliberately so that should steam form within the fuel tubes, the power of the reactor would decrease. As a result of this design, there is an inherent stability in the operation of the N Reactor that is not available in a reactor with a positive void coefficient.

An intense fire at the Chernobyl reactor was caused by the burning of the large graphite stack that was used as a moderator. After the explosion that breached the Chernobyl reactor vault, air was able to circulate through the reactor core and provide oxygen that fed the fire, complicating efforts to bring the plant under control and probably exacerbating the release of radionuclides to the environment. The graphite fire at Chernobyl had no part in initiating the accident. It was a result of the extreme conditions (high temperatures and air ingress) introduced by the preceding, highly destructive thermal explosion. Notwithstanding the problems exacerbated by the Chernobyl fire, the presence of graphite in a reactor system can enhance reactor safety. Graphite has a high heat capacity, which slows the rate of temperature rise in the reactor core, thus mitigating the effects of some potential accidents. Many possible severe accidents in graphite-moderated reactors (but not those involving a reactivity excursion such as that at Chernobyl) would be expected to develop slowly, allowing time for remedial actions that can bring the accident under control.

The Chernobyl reactor also relied solely on slow-moving control rods to control reactivity. Although these depended only upon gravity to insert into the core, they were inserted against a counter flow of coolant so that a full 20 seconds was required for a control rod to move from the full-out to the full-in position. Moreover, the control rod design of the RBMK is such that after the control rods have been fully withdrawn, subsequent insertion causes reactivity first to increase slightly and it only decreases upon further insertion. This initial reactivity addition further exacerbated the reactivity excursion in the Chernobyl accident. By contrast, insertion of the N-Reactor control rods does not lead to an initial increase in reactivity, and the rods are driven in by hydraulic pressure. The time for insertion from the full-out position is less than two seconds—a fraction of that at Chernobyl. In addition, the N

Reactor has a secondary shutdown system of boron carbide balls that in an emergency would drop into the reactor from hoppers located above the core. The boron would absorb neutrons and halt the chain reaction in the fuel. If this mode of shutdown were called upon, the balls would drop to their final positions in less than two seconds. The N Reactor thus has redundant control systems that are much faster than those available in the Chernobyl design.

The containment of an RBMK, called an "accident localization system," consists of several isolatable compartments, each of which is designed to withstand the rupture of the largest pipe in the compartment. This design contrasts with the containment design used in power reactors in Western Europe and the United States. Western-style containment structures surround the reactor system with a single large volume, which is similarly designed to withstand the rupture of the largest pipe within the volume. The accident localization system of an RBMK is intended to limit the release of radioactivity from the plant in the event of an accident by retaining the radioactivity within essentially leak-tight volumes. The accident localization system also prevents the spread of radioactive contamination within the reactor. During the accident at Chernobyl, pipe ruptures were far more extensive than that considered in the design of the compartments of the RBMK accident localization system. As would happen in any containment subjected to loads well in excess of the design, overpressurization and rupture of the containment resulted and led to massive releases of radioactivity from the plant.

The confinement concept employed for the N Reactor (and the Savannah River reactors as well) is quite different than the containment concepts employed either in the Soviet Union or at U.S. commercial plants. In the event of an accident, the confinements deliberately vent the atmospheres surrounding the reactors. The confinements do not then need to withstand prolonged high pressures. The design basis accident assumes that the release of radioactivity will not accompany the early large increase in coolant pressure and evolution of steam, so that the initial venting will not release radioactivity to the environment. When the initial pressure pulse has passed, the vent valves are designed to close based on the reduction in pressure. Attenuation of the ensuing radioactivity release can be accomplished because the venting of the confinements is along pathways equipped to trap much of the radioactive material in the atmosphere. Releases of radioactive noble gases are not ameliorated by the confinement systems, but catastrophic rupture of the confinements by over-pressurization leading to wholesale release of all types of radioactive material is thought by the contractors to be unlikely.

Nevertheless, an important issue remains. Loads that would result from severe accidents at the production reactors could exceed those used as the design basis and the question is whether such loads would cause the confinement function to fail. The need for consideration of potential loads in very severe accidents is discussed in Chapter 2 of this report.

Although the designs of American reactors are quite dissimilar from the RBMK design, there are lessons from the Chernobyl accident that can be applied at all nuclear plants, including the U.S. production reactors. Perhaps the most important lesson is that a severe accident can occur unless design and operation are approached with care and integrity. The Chernobyl design revealed inadequate consideration of severe accidents, and lack of vigilance allowed direct violation of accepted safety practice by plant operators and their supervisors. Another lesson of Chernobyl is the importance of understanding the mechanisms and consequences of severe accidents. Operators must have this understanding or they may induce an accident by violating rules they do not understand. Although the accident mitigation systems and emergency operating procedures normally in use at Chernobyl may have been consistent with the design basis accident for the RMBK reactor, the unavoidable fact is that an accident more severe than the design basis did occur.

POSITIVE SAFETY FEATURES OF THE PRODUCTION REACTORS

In addition to their relatively isolated locations, the Savannah River and N reactors have certain design and operational characteristics that stand out either because of their uniqueness or because they are highly desirable from the point of view of minimizing both the likelihood and the extent of accidents. These features provide a general background to the particular safety issues raised in Chapter 2.

- The Savannah River and N reactors have been operated for over 25 years. There is extensive operating experience with the reactors.
- Both the Savannah River and N reactors operate under negative void and power coefficients (the latter refers to a decrease in reactivity with an increase in reactor power). This makes for more stable operation, simpler control systems, and resilience in responding to increased power conditions should they occur.
- In both types of reactors, the control mechanisms are designed to enter the core in low-pressure regions. At Savannah River, the reactors are operated at low temperatures and pressures. At N Reactor, the control rods are inserted into the graphite stack through cooling channels that are separate from the high-pressure process tubes. This not only facilitates and accelerates insertion of the control mechanisms but largely eliminates the potential for events that could result in accidental ejection of those mechanisms.
- Both types of reactors have backup shutdown systems. Each of the Savannah River reactors has a liquid (gadolinium nitrate) injection system, and the N Reactor has a system of boron carbide balls that makes use of gravity for insertion into the core.
- Both types of reactors have emergency core cooling systems that can be reliably activated. The Savannah River reactors emergency core cooling systems are powered by diesel generators that run continuously, and there is a virtually unlimited supply of cooling water. The N Reactor also has a highly reliable power system for its emergency core cooling system—with steam-driven power generation and diesels—and is further protected by a graphite cooling system (cooling tubes that run through the graphite moderator at right angles to the pressure tubes). For totally unprotected loss-of-coolant accidents, the large heat capacity of N Reactor's large graphite moderator stack would delay any long-term heatup.
- The Savannah River reactors operate at much lower pressure and temperature than either the N Reactor or commercial power reactors. Consequently, their stored energy is significantly lower and any sudden release of energy would be relatively less damaging.

These positive features should be borne in mind as the various issues discussed in the subsequent chapters are examined.

ORGANIZATION OF THE REPORT

The report is organized in three main topics. Chapter 1 examines the overall DOE safety system, and the particular philosophy, objectives, orders, and standards that govern the safety of the production reactors. Chapter 2 addresses a collection of significant safety-related technical issues. Chapter 3 assesses the management structure through which DOE and its contractors pursue safety. Throughout, the committee has sought to recommend actions that, if implemented, can provide a basis for establishing an adequate safety system, for resolving urgent technical issues, and for instituting a management arrangement that leads to independent safety oversight of the defense production reactors.

1
The DOE Safety Framework

Safety results from the firm implementation of sound concepts. Chapters 2 and 3 discuss technical and organizational aspects of implementation; this chapter examines the conceptual foundation of safety at the Department of Energy (DOE) reactors.

With respect to the production reactors, the DOE has established a safety system that consists of three major elements:

- A safety objective,
- Orders that prescribe the means for achieving the objective, and
- A process for ensuring and verifying compliance.

The committee concludes that DOE's conception and implementation of each of these elements is less comprehensive, understandable, and consistent than necessary to assure the continued safety of the defense production reactors.

In addition, DOE production reactors have unique designs, making incorporation of operating experience elsewhere a difficult, and often impossible, task. This imposes a special burden on the DOE safety system, as compared with the commercial industry where there are multiple units of similar design, and requires a high degree of technological excellence in the DOE and contractor staffs. As described later in this report, the level of technological vigilance that these organizations are currently applying to the assurance of safety is inadequate to the task.

THE SAFETY OBJECTIVE

> *Conclusion: The Department of Energy has not clearly articulated, documented, and implemented a safety objective for the operation of its production reactors.*

The design, construction, and early operation of DOE's production reactors preceded the establishment of commercial reactor regulations. Moreover, the production reactors themselves differ significantly from commercial power reactors in purpose, design, and operational characteristics. These considerations have made it impractical to adopt directly regulations established by the Nuclear Regulatory Commission (NRC) for application to the DOE production reactors. Nonetheless, DOE has publicly embraced the general objective of ensuring that DOE nuclear reactors are at least as safe as comparable NRC-licensed nuclear power plants.

Thus, the keystone of DOE's approach is "comparability"—the goal of making DOE reactors at least as safe as commercial nuclear reactors. Comparability is the guiding philosophy for DOE's Orders to its contractors and its procedures for ensuring and verifying compliance. Nonetheless, the committee concludes that comparability is at best an inexact guideline, so that the safety achieved in practice at the DOE reactors cannot be clearly related to commercial regulatory standards. It is important to observe, in this connection, that the inability to compare safety levels does *not* imply that the DOE reactors are unsafe.

The stated DOE objective of comparability with commercial reactors emerged in the 1960s—at the time of the development of commercial reactor licensing. The Atomic Energy Commission (AEC) asked the contractors for the production reactors to conduct safety analyses using the safety philosophy on which the licensing process was based. The purpose was to determine whether modifications were needed in the production reactors to achieve the same level of safety required of commercial-reactor license applicants.

The safety philosophy that came to be used to oversee the production reactors is similar to that used to regulate commercial reactors. The central idea is "design-basis accidents," hypothetical scenarios used to establish design requirements and set operating limits for the plants. These accidents were not defined as the most severe accidents that could be envisioned, but rather were less serious accidents that, although viewed as improbable, were conceivable occurrences. A "single failure" criterion was also used in assessing designs; a safety system was to be available to perform its function even if one active component of the system was assumed to have failed, in addition to whatever failures were assumed to have initiated the postulated accident. The purpose of the single failure criterion was to promote reliability by requiring redundancy and diversity in systems that must mitigate accidents: either two separate and independent systems of each kind or a backup system capable of saving the reactor if the primary system were to fail. The use of these principles in the analysis of the production reactors, years after the plants were constructed, led to the installation of backup safety systems and other safety equipment, which are believed to augment significantly the safety of the reactors.

Today, the objective of comparability is included in DOE Orders, in the Department's contract with Du Pont at Savannah River, and in the contractors' safety analyses of the production reactors, as well as in evaluations by DOE headquarters and the field organizations of contractor performance in selected technical areas. Furthermore, the DOE contractors have explicitly suggested in safety analyses that levels of safety comparable to those attained at commercial reactors have in fact been achieved at the DOE reactors. The DOE reactors are, however, quite different in design and performance from the commercial reactors. In light of those differences, it is extraordinarily difficult to provide a consistent or clear specification at the engineering level of what comparability means, nor of the methods to demonstrate that comparable levels of safety are in fact being achieved. This may explain why the committee received contradictory statements from DOE officials as to whether comparability was actually the objective for these plants. Because of these facts, perhaps the only way to clearly specify comparable levels of safety is to use probabilistic estimates of the frequency of core-damage accidents or the frequency of large offsite releases of radioactivity. But neither of these can be calculated with sufficient confidence of accuracy to be of any actual use to DOE or its contractors in making safety decisions at the production reactors. Thus, although comparability has been a useful tool that over the years has motivated improvements in safety systems in the production reactors, it has not served—and perhaps cannot—as a clear safety benchmark.

DOE has not specified the levels of safety that must be attained, and the concept of comparability is not sufficiently concrete to provide adequate guidance in the absence of such a specification. With no clear objective spelled out, judgments and interpretations are made in a decentralized fashion. This has resulted in the arbitrary and inconsistent application of commercial standards at the two production sites. For example, in order to demonstrate compliance with the commercial reactor limits on radiation doses at the site boundary during accidents, as reflected in the NRC's siting criteria (10 CFR 100), assumptions must be made concerning the hypothetical performance of the production reactor confinement systems in preventing the release of radionuclides as compared to commercial reactor containment systems. (The application of the NRC site criteria to the Savannah River and N reactors is discussed in Appendix H.) These assumptions have not been articulated and examined in a systematic fashion. DOE needs either to clarify the purposes for which 10 CFR 100 is to be used or develop a more meaningful standard for assuring public health and safety.

There are other commercial standards for which the question is not so much how the standards have been applied, but whether they have been applied at all. For example, although DOE Orders require the application of the NRC's general design criteria for modifications to production reactor safety systems, the need to comply with those criteria is contingent upon a determination by DOE that safety would be significantly improved by such compliance. It is unclear what process is to be used to make such a determination. The recent DOE Design Review of the Savannah River reactors compared the reactors against the NRC's requirements and found that the reactors do not meet several important criteria.

In summary, there is no clear explanation of how comparability is or should be used in the design, operation, and safety review of the DOE reactors. The committee's interviews with DOE and contractor personnel raise an even more fundamental concern; those interviews indicate that the objective of attaining comparable safety is neither consistently applied nor fully accepted by DOE and its contractors. The confusion and disagreement over the basic safety objective for the production reactors may help to explain why DOE and its contractors have been slow in developing programs for upgrading these facilities to maintain safety margins, and why they have been slow to address a range of issues related to the potential for severe accidents. Furthermore, the committee was told in interviews that, in the past, the contractors have pressed for safety upgrades that DOE has then rejected for budgetary reasons. In short, the safety objective is so imprecise that it does not provide a means to establish whether improved safety is needed nor does it illuminate how urgently the improvements are required.

The committee does not take issue in principle with the goal of comparability. As the idea is put into concrete form in DOE Orders and in contractor behavior, however, the weaknesses of the philosophy of comparable safety become apparent:

- There are few quantitative standards to use.
- Extensive discretion in the application of commercial standards is left to DOE staff and contractors. That discretion is not subject to explicit guidance.

The committee recognizes, in addition, that objectives other than comparability may be equally valid, including the promulgation of quantitative safety goals, such as those adopted by the NRC. Whatever safety objective is chosen, however, it is essential that a practical and demonstrable system to implement that objective be established. A logical framework of standards and safety performance criteria must be instituted to ensure that the desired level of safety is achieved and maintained. Any objective without such a supporting framework will not provide a consistent basis for decisions concerning production reactor design, modification, and operation, and will, in our view, fall short of public expectations for the safety of DOE's facilities.

> *Recommendation: The Department of Energy should clarify its safety objective for operation of the production reactors. The objective should be operationally meaningful to DOE and contractor staff, and understandable to the public. It should provide a clear foundation on which the implementation of safety can be built.*

DEPARTMENT ORDERS

Conclusion: The Department of Energy has failed to specify

clearly the safety requirements imposed by its Orders, has failed to apply them uniformly at the two reactor sites, and has failed to implement them in a timely manner.

DOE regulates the production reactors by promulgating Orders that, as a result of contractual provisions, must be followed by the contractors. Table 1.1 presents a list of the main safety-related Department Orders that currently apply to the production reactors. (Each of these Orders is briefly described in Appendix H.)

The committee finds significant ambiguity in many of the Orders and a lack of vigorous and timely implementation. In several instances the requirement in an Order refers to an NRC or industry standard but is either so qualified or so ambiguous as to raise doubt as to whether the standard is actually applicable. For example, reactor personnel are required to meet the training and qualification requirements of ANS 3.1 only "to the extent appropriate." The relevant NRC regulatory guide "need only be considered." Plant modifications only have to meet the NRC's general design criteria if and when DOE determines that safety can be "significantly improved." Contractors are required to document technical specifications, but the technical specifications themselves need only be "similar to those required for comparable facilities licensed by the NRC. . . ." Although a DOE Order states that an ANSI/ASME industry standard for quality assurance programs ". . . is the preferred standard for quality assurance," the order elsewhere states that DOE encourages "the judicious and selective application of elements of appropriate, recognized standards . . ." for quality assurance programs.

DOE Order 5480.4 specifically addresses the question of standards. It defines five sets of industry codes as "mandatory environment, safety, and health (ES&H) standards" that DOE contractors are required to meet as a matter of DOE policy, but only if and when DOE determines that their application would increase safety. The committee could find no formal documentation that such a finding had ever been made. In addition, DOE Order 5480.4 lists many "reference ES&H standards," which are described as "references on good practice."[1] But the reference standards are not prescribed for operation of the reactors and are not implemented in a timely fashion by the contractors.

One instance of significant delay in application of standards

TABLE 1.1 DOE Reactor Safety Orders

Order No.	Title	Date Issued or Revised
1300.2	DOE Standards Program	12/18/80
5480.7	Fire Protection of DOE Facilities	12/18/80
5484.1	Environmental Protection, Safety, and Health Protection Information Reporting Requirements	8/13/81
5500.2	Emergency Planning, Preparedness, and Response for Operations	8/13/81
5500.3	Reactor and Nonreactor Nuclear Facility Emergency Planning, Preparedness, and Response Program for DOE Operations	8/13/81
5500.4	Public Affairs Policy and Planning Requirements for Emergencies	8/13/81
5480.4	Environmental Protection, Safety, and Health Protection Standards	5/15/84
5000.3	Unusual Occurrence Reporting System	11/07/84
5480.1B	Environmental Protection, Safety, and Health Protection Program for DOE Operations	9/23/86
5480.6	Safety of DOE-Owned Reactors	9/23/86
5481.1B	Safety Analysis and Review System	9/23/86
5482.1B	Environmental Protection, Safety, and Health Protection Appraisal Program	9/23/86
5700.6B	Quality Assurance	9/23/86

relates to the environmental qualification of reactor electrical equipment—that is, documented assurance that electrical systems are capable of operating for prolonged periods under potential accident and post-accident conditions. Environmental qualification of equipment is a general design criterion for commercial reactors (covered in 10 CFR 50, Appendix A), and was a "recommended" DOE standard as early as 1981. In 1984, DOE issued Order

5480.4, upgrading the environmental qualification requirement to a "mandatory" standard. The contractor at the N Reactor then hired the General Electric Company (GE) to assess the need to environmentally qualify certain systems at N Reactor. GE submitted a report in March 1986 that stated that some systems would need to be qualified to meet the applicable standards. The contractor's current environmental qualification program has been incorporated into the N-Reactor accelerated safety enhancement program, but it will be several years at least before all of the equipment in the plant can be reviewed to determine what needs to be upgraded to meet the standard. Obviously, such delayed application serves to limit the effectiveness of the standard in assuring safety. In this case, however, there has been delay upon delay: a delay of some seven years from the time the NRC passed its environmental qualification rule to the time DOE issued its standard, plus additional delay in implementing the standard after DOE issued it.

The committee found several other indications that DOE does not always vigorously ensure that all of the requirements imposed by its Orders are implemented in a timely manner. In some cases the Orders have not been implemented, apparently because of limited staffing of the field offices. In other cases DOE's extension of a formal waiver has allowed a delay in implementation.

For example, in 1977, the operating contractor for the N Reactor suggested that a better system for the control of liquid effluents was necessary, but DOE did not accept the UNC proposal. However, on March 20, 1984, DOE issued an Order (DOE Order 5820.2) that stated, in effect, that a discharge of liquid effluents to the sand cribs at the N Reactor was unacceptable. DOE directed that:

> Disposal operations involving discharge of liquid Low Level Waste (LLW) directly to the environment or on natural soil columns shall be replaced by other techniques such as solidification prior to disposal or in-place immobilization, unless specifically approved by Heads of Field Organizations, in consultation with [DOE headquarters].

The Order is still not applied to N Reactor, however, because a waiver was granted authorizing continued use of cribs. In light of the $80 million to $100 million cost to upgrade the effluent control system and the limited lifetime of the facility, it is possible that the waiver will be extended throughout the remaining life of the reactor.

Recommendation: The Department of Energy should revise its Orders to specify clearly the requirements imposed and deadlines for implementation. In addition, DOE and Congress must provide adequate funds for implementation.

VERIFICATION OF COMPLIANCE

Conclusion: DOE headquarters has only recently undertaken appraisals of field organization and contractor safety programs that are comprehensive and related to the Department's safety objective.

Prior to the Chernobyl accident, DOE headquarters appraisals of DOE field organization and contractor reactor safety programs were only infrequently conducted. For example, in the six years prior to the Chernobyl accident, there were two headquarters appraisals of reactor safety at Savannah River (in 1982 and in 1986, just before the accident) and only one at Hanford.

The extent of headquarters' involvement with the DOE field organizations and the contractors began to change with the appointment of a new Secretary of Energy in 1985. One of the Secretary's first acts was to request a review of the soundness of the Department's ES&H programs. That review ultimately led to the consolidation of the DOE's ES&H functions under a single Assistant Secretary and to the initiation of what was expected to be a short-term program of headquarters appraisals of the field organizations and their nuclear contractors. These were to be directed by the new Assistant Secretary and her staff. A schedule for the technical safety appraisals was drawn up in the months preceding the Chernobyl accident.

The accident prompted DOE to accelerate its schedule of appraisals and to organize teams of outside experts to conduct additional reviews: separate design reviews of the Savannah River and the N reactors; a special safety review of the confinement and graphite features of the N Reactor; six independent reviews of the overall safety of the N Reactor by a group of outside experts (the so-called Roddis panel); and the review by this committee. Taken together, these reviews represented the first thorough and independent evaluation of the production reactors since the breakup of the AEC.

It is unclear what the future relationship between headquarters and DOE field organization appraisals will be, or the extent to which commercial standards for reactor design and operation will continue to be used to provide criteria for evaluating contractor performance. Adequately sorting out the former will require changes in organizational structure of the kind addressed by the committee in Chapter 3, while the latter cannot be properly addressed until DOE establishes a meaningful safety objective and the supporting standards to implement it.

> *Recommendation: The Department of Energy should conduct comprehensive, high-quality audits of contractor performance, including the use of substantive engineering analyses and rigorous inspection procedures, on a frequent and continuing basis in order to assure compliance with departmental orders.* (Specific recommendations on the organizational structure to enhance the effectiveness of these audits are presented in Chapter 3.)

TECHNOLOGICAL VIGILANCE

> *Conclusion: The age and unique designs of the DOE production reactors demand high levels of technical capability in operating, engineering support, and supervisory staff. Levels of technological vigilance greater than those found in the commercial nuclear industry are needed to reach comparable levels of safety.*

Rules, goals, supervision, and evaluations of performance are necessary but not sufficient to assure safety. The technical competence of workers in both operations and management, in addition to their motivation and sense of responsibility, are essential as well. The committee views these attributes as aspects of technological vigilance. In light of the unique characteristics of the DOE production reactors, high levels of technological vigilance—higher even than those found in the commercial industry—are necessary just to meet the objective of providing comparable safety.

There are two outstanding characteristics that place special demands on the management and oversight of the DOE reactors. First, the production reactors are older than most of the nuclear reactors currently in operation. Modifications to the reactors and their control systems have made the DOE reactors safer than when they were first constructed. But they are now more complex than when they were new, and the complexity is inimical to safety in ways that are not easy to delineate. The aging of components and subsystems increases the likelihood of failures and places special requirements on the reactors in the areas of maintenance and testing.

Second, the uniqueness of the production reactor designs limits the applicability of analyses and experience gained elsewhere in the nuclear industry. For this reason, the experimental and analytical results necessary to determine reactor performance under accident conditions—which are now being developed by DOE contractors—need to be comprehensively and carefully reviewed. The difficulties encountered in verifying the performance of emergency core cooling systems at Savannah River (discussed in Chapter 2) illustrate the problem well.

In the committee's view, DOE and its contractors can only become a standard-bearer in the field of reactor safety through sustained, onsite development of reactor safety technology. In particular, methods and analytical techniques that are at least comparable in technical sophistication to those used in the commercial industry need to be developed and utilized in conjunction with rigorous peer review.

Fostering high levels of technological vigilance at the defense production reactors requires a combination of high degrees of technical expertise and effective management. Some degree of contractor staff expansion may be required to conduct the additional analytical and experimental work needed to assure the safety of the reactors. Moreover, given the overshadowing of DOE personnel by contractor staff at both Savannah River and Hanford, as discussed in Chapter 3, augmentation of the DOE staff is recommended.

> *Recommendation: The Department of Energy and its contractors should develop an expanded in-house capability to address, evaluate, and achieve an in-depth understanding of the technical issues associated with the safety of the production reactors. Methods and analytical techniques that are at least comparable in technical sophistication to those used within the commercial nuclear industry should be developed and used. The analyses and supporting tests needed to support this effort should be*

comprehensive in scope and should be subjected to thorough outside peer review.

NOTE

[1] Prior to 1984, DOE made a distinction between "prescribed" and "recommended" standards similar to the later one between mandatory and reference standards. The earlier distinction was as ambiguous with regard to applicability as the later one.

2

Technical Issues

In the course of its study, the committee identified a number of technical issues that should be resolved in order to improve the safety of the production reactors. These issues, the topic of this chapter, are listed in Table 2.1. Some of the issues relate to design-basis accident considerations, but the majority arise principally from a consideration of more severe reactor accidents—accidents involving disruption and melting of reactor fuel.

The committee recognizes, of course, that in more than 30 years of operation there have been no severe accidents at the defense production reactors. Yet despite the largely satisfactory operating experience of these plants, concerns remain about the possible behavior of production reactor safety systems should an accident occur. The issues are of two kinds. First, there are issues relating to safety equipment, such as whether emergency core cooling or confinement systems would operate as intended in an accident. Second, there are issues relating to operation of the production reactors, and, in particular, whether a severe accident could inadvertently be triggered and whether the staff would respond properly to a severe accident if it were to occur.

Few industries are judged by their ability to handle severe accidents. Commercial aviation, commercial nuclear power, the

TABLE 2.1 Technical Issues Covered in Chapter 2

Acute aging phenomena
Maintenance and plant modernization
Power operating limits
Probabilistic risk evaluation
Severe accident evaluation
Confinement systems
Hydrogen generation and mitigation
Cermet fuel
Human performance
Liquid effluents
Emergency planning

nuclear weapons complex, and, implicitly, defense nuclear materials production are nonetheless held to this standard. In light of the widespread public concern resulting from the Chernobyl accident, and because of its considerable experience in commercial reactor safety regulation, the committee has stressed severe-accident-related issues in its examination of the production reactors. The reader should recognize that a committee with a different make-up could well have focused on different topics. In that connection, the committee wishes to emphasize that attention to severe accidents cannot in and of itself assure the safety of the production reactors and must not be allowed to detract from programs with more immediate benefits—namely, those programs essential to managing, operating, and maintaining the production reactors so that accidents do not occur in the first place.

ACUTE AGING PHENOMENA

Conclusion: The production reactors all display symptoms of acute aging that could affect safety and are likely to limit the useful lives of these reactors.

The Savannah River reactors were built in the 1950s, and the N Reactor began operation in 1963. Both reactor types are old and are beginning to experience life-limiting, material aging brought on by irradiation and corrosion. The contractors operating the production reactors are well aware of these acute aging problems and have introduced a variety of measures to cure or mitigate the aging processes. Summary descriptions of the acute aging problems are presented below. Further details are provided in Appendix D.

Savannah River Reactors

Stress corrosion cracking of the stainless steel piping system and reactor tank is the most acute aging problem facing the Savannah River reactors. To avoid corrosion of the aluminum-clad fuel, the Savannah River reactors operate with a deuterium oxide coolant that is mildly acidified and contains a relatively high level of dissolved oxygen. In this chemical environment, the reactor's high-carbon content stainless steel is susceptible to oxygen-induced, intergranular stress-corrosion cracking in the regions of welds. The significance of cracking is that cracks may be susceptible to unstable growth that could lead to catastrophic rupture of the coolant system.

Cracks were detected in a particularly susceptible "knuckle" region of the C-Reactor tank in 1967. Since then, the Savannah River contractor has upgraded the inspection program at the Savannah River reactors to detect further cracking.

A satisfactory repair process for cracking detected in piping systems is simply to replace the affected piping. Repairing cracks in a reactor tank presents a more challenging problem. In 1968 patches were welded over the cracks in the C-Reactor tank, but leaks were detected in the heat affected zones of these welds in 1984. These leaks are probably the result of the accumulation of helium bubbles in the metal of the reactor tank wall, arising from neutron irradiation of the steel and subsequent radioactive decay to yield helium. Since the welds were at locations on the tank that experienced low neutron fluences and, consequently, had relatively low accumulations of helium, welds susceptible to stress-corrosion cracking in other regions of the reactor tank, where neutron fluences and helium accumulations are higher, would be expected to be even less easily repairable by welding. Replacing the C-Reactor tank has been deemed infeasible. At the end of 1986, in the absence of effective methods to repair the cracks, the C Reactor was retired from service.

The P, K, and L reactor tanks at Savannah River are also susceptible to stress corrosion cracking, although, to date, no such cracking of the tanks has been observed. The Savannah River contractor has developed an operating procedure based on "detect-before-fracture" and "leak-before-break" philosophies. That is, extensive surveillance of susceptible regions of the coolant system is being conducted to identify cracks or flaws. Calculations based on conservative applications of elastic-plastic fracture mechanics are conducted to determine the size beyond which growth would be unstable under normal operating stresses or abnormal emergency conditions.

The committee believes the "detect-before-fracture"/"leak-before-break" operating philosophy for the Savannah River reactors is likely to succeed because the reactors operate at low pressure, with modest stresses on the susceptible regions even under abnormal conditions. (This is also the judgment of outside consultants hired by the contractor.) The contractor is handicapped in applying this procedure because only visual identification of cracking in the vessel can now be performed. This is because of the unique design of the Savannah River reactors, which does not lend itself to the ready use of ultrasonic testing equipment that has been designed for use in light-water reactors. The committee believes that the contractor's efforts to develop ultrasonic crack detection methods suited to the Savannah River reactors are to be encouraged, as are the contractor's continued efforts to develop technology for repairing cracks in the reactor tanks. Without such repair technology, all of the Savannah River reactors may eventually have to be retired from service due to stress corrosion cracking.

N Reactor

Two acute aging problems beset N Reactor. Both are brought on by the prolonged irradiation of reactor materials over the life of the reactor. Radiation damage is causing the high fluence regions of the graphite moderator to expand. Because the expansion is nonuniform, it places stresses on the process tubes and the graphite cooling tubes interspersed throughout the graphite moderator. The fracture of four cooling tubes has been attributed to stresses on the tubes caused by the offsetting movement of adjacent stacks of graphite. The expansion of the graphite is also distorting the horizontal control rod channels and the vertical channels for the boron carbide ball scram system. At the same time, the process tubes are being embrittled by exposure to radiation. The embrittlement is augmented by zirconium hydride formation, which enhances stress at the location of cracks, making the metal less fracture resistant. Gouging of the distorted process tubes during fuel insertion provides opportunities for crack formation and subsequent failure of the tubes. In sum, graphite expansion is placing stresses on hardware within the N-Reactor core at a time when irradiation-induced embrittlement is making the metal components that reside for long periods less able to withstand these stresses.

Expansion of the graphite will reach a critical juncture when the moderator comes into contact with the reactor vault's biological shields. The Department of Energy has indicated that when this happens—and it is expected to occur sometime between 1991 and 1996—N Reactor will be retired.

In the meantime, the contractor operating N Reactor is developing a program to monitor the graphite growth and to assess the deterioration of the fracture toughness of the hardware in the core. Although the procedures that the contractor intends to use to monitor process-tube embrittlement seem well founded, the committee questions whether the application of these procedures is sufficiently extensive. Initial plans called for the destructive, metallurgical and strength testing of only 4 of the 1003 process tubes in the N-Reactor core, and nondestructive testing of only about 5 percent of the tubes. There are four distinct types of tubes in the reactor core, resulting from the use of three different manufacturers and two levels of cold work on one manufacturer's tubes. Radiation effects on the tubes vary from location to location as well as varying with the method of tube manufacture. As a result, the initial surveillance plan was almost certainly too limited to yield a proper statistical representation of multivariate effects. Recently, indications of surface and subsurface flaws in N-Reactor process tubes have been found using eddy-current and ultrasonic techniques. This has resulted in extending the number of nondestructive examinations to 152 tubes, including complete examinations of all tubes in the class of tubes with the most numerous indications of flaws. Comparison of results obtained with nondestructive techniques to results obtained by destructive examinations indicates that the nondestructive techniques yield low estimates of the length and depth of larger flaws in the tubes. As a result, the contractor has decided (1) to remove any tube which by nondestructive methods is indicated to have a flaw deeper than 0.1 inch (the process tube wall thickness is 0.275 inches), (2) use analyses to assess the growth rate and acceptability of flaws less than 0.1 inch deep, (3) remove a total of six process tubes for destructive examination, and(4) continue to update and improve nondestructive examination techniques and analysis models.

The contractor does not now plan 100 percent nondestructive examination of the process tubes. The committee believes that 100 percent examination may be worthy of consideration in light of the frequency of flaw indications in the process tubes examined to date, uncertainties in the reliability of nondestructive techniques for indicating flaw sizes, and uncertainties in the growth and behavior of flaws.

The contractor has also suggested a variety of methods both to ameliorate the effects of graphite expansion and to slow the rate at which it is occurring. The expansion rate can be slowed by flattening the neutron flux distribution throughout the N-Reactor core so that radiation damage is accumulated at a slower rate across a greater volume of graphite. This will, of course, involve changes in the neutronics of N-Reactor operation and will require extensive analysis (for example, of xenon oscillation effects) prior to implementation. The contractor is aware that flattening the flux distribution increases the power density in the outer portions of the core where the graphite cooling system is least effective. For this reason, the contractor is currently assessing the implications of the flux-flattening program for the analysis of those severe accidents that are mitigated only by effective graphite cooling.

Slowing the rate of growth of the graphite moderator will not preclude a range of potential problems. Enhanced methods of in-service inspection of the least accessible regions of the graphite will be increasingly important in detecting any unexpected local distortions or graphite damage.

The expansion of the graphite is also distorting the horizontal control rod channels and the vertical channels for the boron carbide ball scram system. The committee carefully reviewed these effects. The backup boron carbide ball scram system has an excellent margin for reliable operation in the face of the distortion of the channels, and the contractor has taken steps to assure that the functioning of the control rods is not impaired by the graphite distortion.

Summary

The production reactors are old facilities and despite a variety of costly upgrades that have been implemented over the years, a number of acute problems related to their material aging cannot be avoided. The need to address these problems requires substantial allocation of contractor and DOE resources. One source of the current difficulties is the lack of planning in the past to provide for a safe, continuing production capacity. Within the coming decade that failure could result in the nation having to rely upon a small number of aged production reactors that demonstrate serious safety problems, but with no other facilities ready to take their place to meet the nation's security needs.

Recommendation: The remaining useful life of the four production reactors is likely to be equal to or shorter than the time needed to authorize, fund, design, and build new facilities to produce special nuclear materials, particularly in light of the status of the N Reactor. If the United States finds it necessary to have a reliable and safe capability for the production of strategic nuclear materials, then planning for new production reactors or other alternatives should be accelerated.

MAINTENANCE AND PLANT MODERNIZATION

Conclusion: The increasing needs for maintenance and plant modernization brought on by the aging of the production reactors are not being met by the existing programs.

Programs of corrective and preventive maintenance at both N Reactor and Savannah River appear in need of comprehensive upgrading in order to ensure that maintenance of the production reactors is adequate. The need for improved maintenance planning, procedures, and systems and better crafts training of maintenance personnel was identified in several post-Chernobyl reviews of the production reactors. These recent calls for improved maintenance echo quite similar comments on production reactor maintenance made following the accident at Three Mile Island, notwithstanding substantial progress since then.

Two philosophies for the maintenance of nuclear facilities are widely recognized. In one philosophy, maintenance is devoted to the repair or replacement of systems and components that have failed or that have fallen in performance below some minimum acceptable standard. A somewhat more sophisticated application of this maintenance philosophy uses projection techniques to forecast minimal operability conditions in order to guide the maintenance effort. The second, far more conservative philosophy of maintenance is based on continually reestablishing the safety and operational margins of systems and components whether or not these have become demonstrably defective. That is, the original performance margins of the plant, rather than minimal operating criteria, provide standards for maintenance. Because nuclear plants are subject to complex interactions between and among plant systems, and because of the ongoing evolution in nuclear facility safety standards, this second, more conservative philosophy is particularly suited to aged facilities such as the production reactors.

Over the last several years, DOE's contractors have formulated plans for upgrading the Savannah River reactors and the N Reactor. All of these plans are of relatively recent vintage and were formulated to remedy problems that developed during a protracted period when little of the budget was available for maintenance and modernization. The most important of these plans are as follows:

1. Productivity Retention Program for N Reactor.

Initiated in 1984, this program is to be complete in 1992. It involves primarily rehabilitation of pumps, valves, and boilers and the upgrading of obsolete instrumentation.

2. Productivity Assurance Program for N Reactor.

The program plan for this work, written in 1985, addresses extension of the useful life of N Reactor. Now part of the Accelerated Safety Enhancement Program, it consists primarily of a surveillance effort to monitor aging phenomena, such as process tube embrittlement and graphite expansion.

3. Savannah River Facilities Upgrade Study.

This program was the forerunner of, and has a very similar thrust to, the Productivity Retention Program at N Reactor.

4. Improved Confinement Facilities for SRP.

This program is an effort to upgrade the confinement at the Savannah River reactors. The principal focus of the project is to enhance the noble-gas retention capability of the confinement.

Though the stated motivation for the above plans is the reestablishment of plant safety margins, it is clear that these plans have been formulated within the context of the first philosophy of maintenance—correction of demonstrably defective components. Funding limitations have prevented these projects from constituting comprehensive, systematic maintenance and modernization programs. For example, in the case of the Savannah River reactors the funding requested for the facilities upgrade initiative amounted to only 1 percent of the estimated replacement cost of the facilities. Consequently, as noted explicitly in the plan, only the most urgent maintenance and upgrading needs were addressed. Similarly, the plan developed for the N Reactor explicitly notes that it addresses only systems currently experiencing failures or projected to degrade seriously by the mid-1990s.

Extraordinary maintenance efforts are needed at the production reactors. The number of unresolved reactor incidents at the Savannah River reactors has been steadily increasing over time, as shown in Figure 2.1. In 1983, there were 130 such unresolved incidents and by 1986 the number had risen to over 250. The number of so-called recurring incidents (an incident that happens at a rate in excess of 8 over a two-year period) has also increased.

One of the most striking examples of the need to modernize is shown by the refueling practices at N Reactor. During refueling workers are routinely sprayed with radioactively contaminated water. Though the workers are equipped with protective clothing, the refueling procedures in use are antiquated and should have been discontinued years ago. Since 1983 approximately $6 million has been allocated for the development of remote, automated refueling equipment. Because of several false starts at engineering the equipment, alternative means of refueling the reactor are still not available. That remote, automated refueling equipment has not been developed and incorporated in a timely manner is symptomatic of the more general failure to modernize properly the N Reactor.

Over the years the Savannah River contractor has developed engineering responses to a range of maintenance and modernization needs identified at the plant—from surveillance programs that respond to the discovery of cracking problems, to the development of computer systems for automatic backup shutdown capability. The committee found, however, that the process employed in designing, funding, prioritizing, and engineering these activities leads

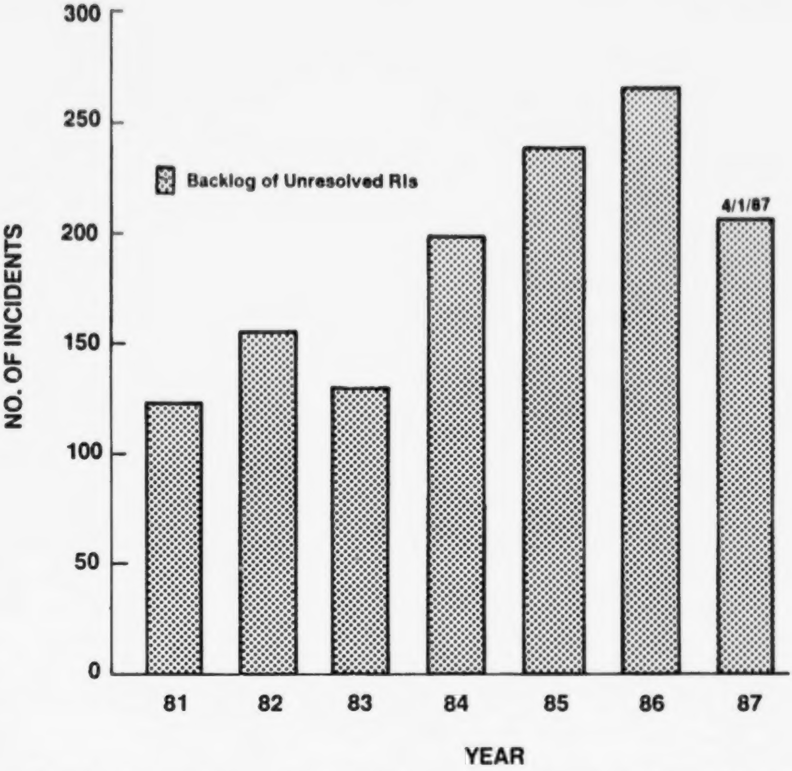

FIGURE 2.1 Backlog of unresolved reactor incidents (RIs). Reactor incidents at Savannah River are classified by Du Pont into five categories on the basis of their safety significance: (A) "incidents with serious consequence" (none of these has occurred); (B) "incidents with significant consequence or hazard potential"; (C) "incidents with remote hazard potential"; (D) "conditionally significant incidents" (which may indicate generic or recurring problems or design deficiencies); and (E) "incidents with no safety potential." The backlog of unresolved reactor incidents plotted in this graph involves incidents in categories B, C, and D. From 1981 through 1985, the distribution of these incidents remained approximately the same (12% Bs, 32% Cs, and 56% Ds).

to long delays in implementation. For instance, unwanted control rod motion caused by obsolete vacuum-tube amplifiers in control rod mechanisms is a problem that has been recognized for years, but remains uncorrected. More generally, the current backlog of unimplemented capital equipment projects at Savannah River includes some that date back more than five years, and schedules for completing recently proposed projects extend into the mid-1990s.

In 1986 the Reactor Safety Advisory Committee for the Savannah River reactors recommended that the backlog of maintenance activities be corrected before the problem became unmanageable. Over the first three months of 1987, extraordinary measures were introduced at Savannah River to reduce the maintenance backlog and to prevent the maintenance workload from growing faster than the work could be done. These efforts reduced the number of unresolved reactor incidents, though the number is still in excess of 200. The current standdown of the N Reactor has been similarly useful in permitting completion of outstanding maintenance work at Hanford.

Maintenance and modernization efforts, as described in the plans for N Reactor and Savannah River, appear to be directed principally at ensuring continued operation of the plants. Although the plans do cite public safety as one motivation for the work, it is unclear how public risk reduction or risk aversion is factored into the setting of priorities for the work. Cost-benefit analyses of modernization efforts at N Reactor currently consider loss of production and loss of revenues from steam sales. While risk of the loss of operation and the risk of radiation exposure to workers are considered, public risk posed by a poorly maintained and modernized plant does not enter explicitly into the computations. Public risk ought to be an explicit factor in making decisions about modifying the plants.

> *Recommendation: Maintenance planning, procedure improvements, and system training should emphasize the special problems inherent in operation of these aging reactors. DOE and its contractors should undertake systematic analysis of aging with increased emphasis on preventive maintenance and replacement of old components with new technology, where appropriate, in order to forestall in-service failures and to reestablish the safety margins of the facilities. Avoidance or reduction of risk should be explicitly considered in the identification and prioritization of maintenance and modernization efforts.*

POWER OPERATING LIMITS

Savannah River

Conclusion: Based on preliminary analysis of the significant power derating that was put into effect during the course of this study, DOE and the contractor believe there is a reasonable expectation that the emergency core cooling system will prevent core damage in the hypothetical case of a loss-of-coolant accident. However, the potential for and consequences of early core heatup caused by flow instability in the first few seconds of the accident remain to be explored.

Conclusion: Adequate resources have only recently been devoted to developing a thoroughly documented understanding of the behavior of the reactors in a loss-of-coolant accident.

In 1979 the Savannah River contractor obtained experimental data in an attempt to quantify the power levels that might lead to dryout (loss of cooling capability in fuel assemblies) during operation of the emergency core cooling system (ECCS). By 1981 the contractor had come to recognize that the 1979 experiments were suspect with regard to adequately representing certain important physical phenomena. Between 1981 and 1986 staff of the Savannah River Laboratory engaged in reviewing the work and conducting new experiments. This work established that during a loss-of-coolant accident, dryout might occur at significantly lower power levels than previously believed. In late November 1986, the contractor notified DOE of the uncertainties associated with the effectiveness of the ECCS, and thereafter the allowable upper limits on operating power of the three Savannah River reactors were lowered by approximately 20 percent.

In February and March of 1987, the committee reviewed the available evidence and notified the Secretary of Energy that there were insufficient technical data to demonstrate that the ECCS performance of the Savannah River reactors could be assured, even at the reduced power levels (see Appendix C). Based on this report and subsequent reviews, the Department further reduced the power limits of the reactors to approximately 50 percent of the original level.

Dryout in the Savannah River reactors is controlled by countercurrent flow limitations (CCFL) in the coolant channels. This can be explained as follows. Normal coolant flow in the Savannah River reactor cores is downward. During a loss-of-coolant accident—a term of art that refers to a design-basis accident in which the largest pipe that carries cooling water is assumed to break—the coolant flow is reduced. For a given low rate of flow, the decay power level can be high enough that it leads to steam production that in turn can interfere with the downward flow of coolant through the core. The recent data indicate that the power level at which CCFL occurs may not be very far above the point of net vapor production (the point where the power is sufficient to heat the water to its boiling temperature).

The problem of CCFL is complicated by the magnitude of boiling and condensation effects in the small coolant channels of the Savannah River fuel assemblies. In addition the parallel channel configuration of the assemblies is a critical factor in .. Although CCFL phenomena have been studied in a variety of different settings and for both nuclear and nonnuclear applications, the geometry of Savannah River fuel assemblies is sufficiently distinctive that the applicability of this past work is severely limited.

Even the most recent experiments at Savannah River are not fully prototypic because the geometry of the test rig is different from the geometry of the full assemblies. This shortcoming has required the establishment of power limits that are low enough to prevent any bulk boiling during a loss-of-coolant accident. The viability of this approach is still contingent upon a quantitative understanding of the flow distribution among the assemblies and within them—that is, whether, under the conditions of interest, all channels receive an adequate amount of coolant from the ECCS.

An additional complication is the need to show that smooth (quasistatic) emergency coolant flow will be quickly established following the flow instability and associated boiling of the liquid coolant in the core within the first few seconds of a postulated loss-of-coolant accident. These early phenomena during a loss-of-coolant accident are a consequence of depressurization of the inlet plenum, which causes the reactor coolant to flash to steam within the fuel assemblies, leading to two-phase (liquid and gas) flow in the core. The two-phase flow exhibits a much larger hydraulic resistance to flow than single-phase (water) coolant. Consequently, subsequent flow results in flow instability, which is exacerbated by the flow decaying faster than the power in the first few seconds of the accident. Here, too, parallel channel effects that may cause coolant to bypass individual channels are critical to an understanding of the reactor behavior. At present, for the Savannah River loss-of-coolant accident, there is an inadequate understanding of coolant boiling, heatup, and flow dynamics.

The Savannah River contractor has recently developed an extensive program to modify the reactors and to improve the experimental and calculational basis for demonstrating the adequacy of system response to a loss-of-coolant accident. The program includes addition of a fourth emergency cooling feed line in all three Savannah River reactors, modifications to the fuel and target assemblies, and various improvements in the experimental and computational data base to support the analysis of loss-of-cooling conditions. The contractor plans to install the fourth additional system in the L Reactor by November 1987 and in the K and P reactors by November 1988. A summary of the contractor's program to restore full power operation is included in Appendix C.

Recommendation: The Department of Energy should ensure that before restoring full power operation at Savannah River it satisfies itself, on the basis of a rigorous external review, that it has a thorough understanding of the behavior of the Savannah River reactors in a loss-of-coolant accident.

N Reactor

Conclusion: A thorough understanding of the behavior of N Reactor in a major loss-of-coolant accident does not exist. Current understanding is based on the dated evaluations reflected in the Safety Analysis Report of 1976. Several past attempts to apply state-of-the-art tools have failed, largely because of numerical difficulties that are now being resolved.

Emergency core cooling system capability associated with a variety of loss-of-coolant accidents was reassessed in the 1976 edition of the N Reactor Safety Analysis Report. A slightly modified version of a computer code developed for application to commercial reactors (RELAP4) was used in these assessments. The most severe case resulted in computed fuel temperatures of approximately 1400°F, at the time that coolant reflood (recovery of core flow) begins. This is only about 400°F below the temperature at which fuel failure would be expected.

Core heating in the N Reactor during a loss-of-coolant accident is determined by a number of factors, including (1) the time required for depressurization (blowdown) to the pressure level at which the ECCS can be actuated and (2) the flow pattern of coolant in the core during this time. The most damaging breaks in the system would be expected to occur in the cold leg of the primary coolant loop. This is especially true for break sizes that result in an early flow stagnation (loss of flow because of closure of check valves in the cold-leg riser). In these cases, the blowdown time would be prolonged, and the core temperature would reach high levels.

This physically intuitive view is supported by the RELAP4 computer code results, but there is a need to reassess the cooling capability over a range of accident scenarios, break sizes, and locations. There are also some special aspects of the calculation that require further evaluation.

1. The RELAP4 code is based on a simplified treatment of two-phase flow (homogeneous equilibrium). This may be adequate during the early stages of blowdown, but this approximation becomes progressively inappropriate as the system empties and loses pressure. The impact of this inaccuracy has not been quantified for the N-Reactor piping system. A 1964 test has been cited as a benchmark for validation of the computer code, but the measurements of transient pressures and blowdown rates in that test contained large uncertainties. The value of the test as a benchmark is further in question because it was an adiabatic test (that is, no heat transfer occurred because there was no secondary coolant and no nuclear power in the core).

2. The RELAP4 code, as used in these calculations, had additional simplifications that are not present in more up-to-date computer codes. For example the whole reactor core was treated as a single flow channel. Similar simplifications were also employed in representing other volumes in the system. This approach can lead to analytical errors in predicting conditions in localized regions and consequently to errors in blowdown rates. Further uncertainty is introduced by the use of an empirically determined discharge coefficient that is constant in time. The validation experiments performed for the RELAP codes are more applicable to commercial reactors than to the distinctive horizontal geometry of the N-Reactor core. As a result, the time-dependent parameters such as

discharge coefficients may not provide a good characterization of N-Reactor depressurization.

3. In many of the calculations the predicted core flow is highly oscillatory. It was initially thought that this result was a consequence of the particular numerical representation of check valves in the cold-leg riser, but more refined calculations failed to resolve this apparent numerical difficulty. It is unlikely that the existence of local numerical instabilities in the analysis would necessarily invalidate the entire blowdown calculation, but the difficulties may be based on real phenomena and should be resolved. Furthermore, if these calculated oscillations are only resolved through numerical means, unrealistically high predictions of core cooling rates may result. The reported results of the calculations were not sufficiently detailed to allow the committee to assess this aspect of the problem.

4. The reported calculations did not analyze the postulated accident into the period of reflooding, and some calculations were stopped even earlier, apparently because of numerical instabilities. Preliminary attempts, using an advanced version of the same computer code (RELAP5), also exhibited instabilities. This difficulty has only recently been overcome and studies are continuing. It is important to recognize that there are physical reasons for expecting condensation-driven flow and pressure instabilities to occur. In fact, an indication of these phenomena was observed in venturi meter data acquired from the hot dump test of 1964. These instabilities must be examined even though they are difficult not only because of the need to develop an understanding of the overall hydrodynamic behavior of N Reactor, but also because they raise the possibility of intense water hammer events which could affect the structural integrity of N-Reactor pressure tubes.

The safety analyses performed for hypothetical loss-of-coolant accidents in the N Reactor have focused on large break accidents because the contractor believes these present the most severe challenge to the emergency core cooling and confinement systems. Since small-break loss-of-coolant accidents are much more likely to occur, it is important to assure that adequate protection is provided for these accidents as well. In the event of a small break in the reactor coolant piping, valves are designed to open, which would depressurize the system, in effect turning small-break accidents into large-break accidents. The committee believes that special consideration should be given to small-break accidents in the ongoing probabilistic risk assessment (PRA) in order to evaluate the likelihood of multiple failures which would prevent depressurization and to estimate the potential consequences of these types of events.

Recommendation: A thorough understanding of the behavior of N Reactor in a major loss-of-coolant accident should be urgently developed. This understanding should be based on state-of-the-art analytical tools, and these tools should have the ability to analyze potential thermal shock and water hammer phenomena. The use of these tools should be accompanied by rigorous quality assurance of the computer codes, as well as experimental validation of the results, particularly with respect to the distinctive horizontal geometry of the N-Reactor core.

PROBABILISTIC RISK ASSESSMENT

Conclusion: The risks associated with operation of the defense production reactors are currently inadequately understood; efforts to evaluate those risks by probabilistic risk assessment methods are still in their early stages.

The DOE has been slow to undertake PRAs for the production reactors. The committee finds this fact disturbing, in light of the time that has passed (12 years) since the publication of the NRC's Reactor Safety Study. Additionally, more than a dozen utility-sponsored PRAs have been completed on commercial reactors while the DOE and its contractors have only recently begun studies for DOE reactors. The principal value of the PRA lies not in the calculation of bottom-line probabilities of severe accidents but in the acquisition of engineering insights into ways to improve plant safety.

PRA studies are frequently divided into three levels of detail and completeness. Level 1 PRAs are limited to calculating the probabilities of accidents that involve significant fuel damage. Level 2 PRAs also include estimates of the timing, type, and amount of radioactive materials released from the plant. Level 3 PRAs include calculations of risks to public health and of economic consequences of radionuclide release.

The general methods of analysis used to perform a Level 1 PRA are independent of the type of reactor being analyzed and have become well developed over the past dozen years. However, the prediction of severe accident behavior, as required to perform a Level 2 PRA, involves methods of analysis that are plant-design dependent. As discussed in the section on severe accident analysis, these methods are not well advanced for the production reactors. Although a Level 2 PRA can be performed with crude methods of analysis, the uncertainties in the estimated risk will be large. Nevertheless, the results can be used as an aid to decisionmaking if the uncertainties are characterized and considered. The extension of a Level 2 PRA to a Level 3 PRA does not involve substantial additional effort.

At the request of the DOE, Los Alamos National Laboratory is undertaking Level 1 PRAs for the production reactors at both sites. Preliminary results for the N Reactor were provided to the committee. These studies are not sufficiently advanced to use as a basis for decisionmaking at this time.

Additionally, both contractors have agreed to perform Level 3 PRAs for their plants. Each is currently performing a Level 1 PRA and developing the methods for a Level 2 analysis. Because the development of the Level 2 methods will require substantial time and will delay the completion of the PRA, the committee suggests that, as a parallel effort to the computer-code based approach that is now being pursued, Level 2 analyses should be performed with simple physically based models. This would permit early insights to be obtained and would provide a physical understanding of the severe accident processes that must be included in the code analyses.

The committee strongly supports the PRA efforts that are being undertaken for the production reactors. We encourage the completion of these studies as expeditiously as practical. The committee also believes, however, that the quality of the PRAs must be high in order to aid in decisionmaking. Some aspects of a state-of-the-art PRA that are essential to a credible product include the following:

- Independent peer review,
- Use of site- and plant-specific data,
- Extensive involvement of plant operators,

- Analysis of external accident-initiating events (e.g., earthquakes),
- Analysis of common-cause failures, and
- Characterization of the uncertainties in the estimated risk.

Recommendation: The Level 1 and Level 2 PRAs currently under way for the Hanford and Savannah River reactors should be completed and subjected to peer review as expeditiously as possible. Their results should be used to evaluate plant modifications that can reduce the assessed risks and to identify and evaluate research to reduce uncertainties. DOE and its contractors should recognize, however, that the principal benefits of the production reactor PRAs will not derive from the accident probabilities that are calculated, but from the engineering insights that normally accompany the development of these tools on a plant-specific basis.

SEVERE ACCIDENT EVALUATION

Conclusion: The existing level of understanding of severe accident behavior for the production reactors is inadequate to permit a realistic assessment of the effectiveness of these designs in mitigating the consequences of severe accidents.

Information on the severe accident behavior of the production reactors is critical. The Administration and the Congress face important decisions about the future of the production reactors which will be difficult to make in the absence of essential information regarding the specific risks of these plants and their effectiveness in mitigating severe accidents. Level 1 probabilistic safety studies of the plants have only recently been undertaken. Extending these probabilistic safety studies to an analysis of public health consequences (the so-called Level 3 PRA) will require a capability to analyze severe accident behavior. As noted, such a capability does not now exist.

Severe Accident Behavior

Improvements were made at the production reactors in operator training and control room operations as a result of the lessons learned from the accident at Three Mile Island. Similar improvements were made in the commercial reactor industry, but in the commercial industry these improvements were complemented by major research efforts to understand commercial power reactors under hypothetical severe accident conditions. Programs to understand hypothetical severe accidents at the production reactors took far longer to get organized and were on a much smaller scale. Prior to Chernobyl, DOE and its contractors continued to rely principally on design-basis accident analyses performed in the late 1960s and early 1970s. These analyses focused on exceedingly unlikely, large-break loss-of-coolant accidents which were considered to be "worst case" or so-called "maximum credible accidents." The attention given these accidents by the DOE contractors is similar to the attention given them by the commercial reactor industry prior to the introduction of more comprehensive probabilistic risk assessment techniques. As a result, the production reactors have been designed, constructed, and modified in order to cope with the events accompanying these "worst case" accidents. DOE and its contractors have given much less attention, however, to more probable accidents such as those involving small breaks in the coolant system or accidents involving multiple or "common mode" equipment failures. In sum, throughout the early 1980s the level of understanding of severe accident phenomena for the production reactors remained roughly comparable to that available to the commercial reactor industry in 1975.

The Chernobyl accident reinforced the principal lesson of Three Mile Island that accidents more severe than the design basis—accidents that were often considered "incredible" in the past—could indeed occur. Although a limited external review of the N Reactor in 1982 and the ongoing response to Three Mile Island at Savannah River in 1984 had engendered programs to reassess the potential for severe accidents at the production reactors, these initiatives were small in scale and low in priority. Since Chernobyl, these programs have been expanded and accelerated.

The program at N Reactor is an exclusively analytical effort initiated in October 1986. Some program elements are to continue through 1989, but most of them are to be completed in 1987. Analyses of severe accidents are to be performed using older computer codes that were developed for the production reactors and with modified codes developed for analyses of commercial reactors. The program is to include reexamination of hydrogen generation and hydrogen behavior, estimation of fission-product release from degrading fuel and the subsequent behavior of the fission products, examination of accidents involving events other than rupture of main coolant pipes, and more up-to-date analyses of accidents considered in the N-Reactor safety analysis report. There are, in addition, analytic efforts in support of ongoing probabilistic safety studies and analyses of reactivity transients and neutronic effects.

The program now being planned at Savannah River is a more comprehensive effort. Current plans call for a program lasting about four years. Major elements in the plan include development of an integrated computer code system to model the progression of hypothetical severe accidents in the Savannah River reactors. There are also plans to develop information and data to support these modeling efforts, particularly in the areas of two-phase flow phenomena, radionuclide release and transport, energetic events such as hydrogen combustion, fuel-coolant interactions and interactions between core debris and concrete, reactor operability, and equipment response to severe accident conditions. This program complements an ongoing effort to upgrade the confinements at Savannah River.

Although the analysis of severe reactor accidents at the production reactors should utilize, wherever possible, the substantial technology developed for commercial reactor safety studies, it is important to recognize that the production reactors are quite different from commercial reactors. As a result, the production reactors would be expected to behave differently under severe accident conditions. The application of computer codes developed for commercial power reactors to the unique circumstances of the production reactors, therefore, presents major difficulties. Experimental data that reflect the unique features of the production reactors are necessary to validate the models and computer codes being used to analyze potential severe accidents but are currently unavailable.

There have been no severe accidents at the production reactors. Still, such accidents cannot be ruled out. A 1985 review of the Savannah River power system, conducted at a time when construction activities then in progress affected the availability of the system, revealed extraordinarily high probabilities for complete loss of power to the reactor areas. The committee does not know the current reliability of the power system, but the earlier report is sufficiently extraordinary to raise concerns. The consequences of the loss of reactor power—station blackout—is being considered in the contractor's PRA.

Areas of severe accident behavior for the production reactors that need investigation include all the phenomena known to be important to an understanding of hypothetical severe accidents in commercial reactors. Among the critical areas that require examination are the following:

1. *Thermal Hydraulics* The behavior of coolant flows in the production reactors is discussed elsewhere in this chapter in connection with the discussion of limits on maximum operating power. Similar uncertainties in the behavior of coolant flows exist with respect to the more drastic conditions that would occur during severe accidents.
2. *Thermal Shock* A related phenomenon of concern is thermal shock. The committee believes that the potential for brittle shock fracture of overheated process tubes during degraded emergency core cooling injection needs to be more thoroughly considered for the N Reactor, especially in light of the age of many of these tubes and the existing uncertainties about the extent of flaws and cracking.
3. *Radionuclide Release* Assurance that the confinement systems of the production reactors will sufficiently mitigate the radiological consequences of a severe accident cannot be provided without knowing the extent and nature of radionuclide release from the fuel. Yet safety analyses of the production reactors have been based on antiquated estimates of radionuclide release that the little experimental data available have shown to be invalid. More recently, in order to accommodate the latest data on radionuclide release from Savannah River fuel, the Savannah River contractor has changed the mode of operation of the Savannah River reactors, placing the emergency spray systems on continuous standby. The committee believes that while this is a useful measure, more thorough, integrated analyses need to be done in order to assure that sprays in the reactor room will sufficiently attenuate the more extensive radionuclide release it is now believed could take place if a severe accident were to occur at the Savannah River reactors.
4. *Fuel Damage Progression* The evaluation of fuel melting in a severe accident raises a number of complex questions relating to the progression of melting and the behavior of the melted material. The data bases available on melt progression

for the production reactors are quite limited, and data obtained for commercial reactor fuels are wholly inapplicable. It has been established that uncertainties in the melt progression phase of a hypothetical severe accident affect predictions both of hydrogen generation for the N Reactor and of core debris interactions after expulsion from the Savannah River reactor vessels. The predicted course of an accident at N Reactor would be radically altered if the assumptions made concerning melt interactions with process tubes are inaccurate.

5. *Hydrogen Generation* Hydrogen generation has proven to be a major issue in the analysis of severe accidents in commercial power reactors. The principal source of hydrogen in commercial reactors (with oxide fuel) is steam oxidation of fuel cladding. In the production reactors not only will the cladding react with steam but the fuel itself will react to form hydrogen. At N Reactor, the graphite moderator block can also react with steam to form both hydrogen and carbon monoxide. Despite these many potential sources of hydrogen, analyses done to date for the production reactors have indicated that little hydrogen would be produced. These results have been criticized in several previous reviews of the N Reactor. The committee also has questions about these findings and believes a more definitive validation of the calculations on which they are based is warranted. (See the discussion on hydrogen generation elsewhere in this chapter and in Appendix G.)

6. *Radionuclide Transport and Behavior in Containment* The accident analyses performed for the production reactors have largely ignored radionuclide deposition within the reactor coolant system and have assumed about a 50 percent deposition of iodine within the confinement. Analyses for the N Reactor have assumed a very high decontamination of evolved gases by the confinement sprays. These assumptions have not yet been verified. By adopting these assumptions, the contractors have obviated the need to analyze possible delayed revaporization of deposited radionuclides that might result in the delayed release of radioactivity to the environment if revaporization were to occur at a time when confinement capabilities are degraded. These are issues that have recently assumed some importance in the analysis of hypothetical severe accidents in commercial power reactors and deserve more thorough consideration for the production reactors.

7. *Behavior of Core Debris Once It Leaves the Reactor Vessel*

The uninterrupted progression of a severe accident at the Savannah River reactors would result in debris being expelled from the reactor vessel. Understanding the ensuing interactions between core debris and concrete has become important in safety analyses of commercial power reactors because of the potential for generating combustible gas and radionuclide release. At the Savannah River reactors, available data do not adequately address the question of interactions between concrete and the unique core debris mass that would be expected to be formed in a severe accident. Potential interactions between reactor fuel and coolant once the fuel leaves the reactor vessel is another issue that deserves more refined analysis at Savannah River. These interactions might be highly energetic, if not explosive, in nature and even nonexplosive interactions could serve as sources of combustible gas, confinement overpressurization, and radionuclide release.

8. *Confinement Loading* The confinement structures surrounding the production reactors were not designed to be robust like the containments surrounding commercial power reactors. The committee is concerned that the production reactor confinements may be vulnerable to rapid pressure pulses. Such pressure pulses need not be shock waves such as those produced by combustible gas detonation or fuel-coolant interaction. Pressure pulses of only a few pounds per square inch could result in rupture of the confinement filters and the direct discharge of fission products from the plant (see the discussion on confinement elsewhere in this chapter). Such pressure pulses could arise as a result of hydrogen combustion or interactions between molten fuel and coolant.

The committee believes that a substantial and time-consuming effort would be required to conduct a severe accident research program that fully responds with the necessary high degree of credibility to the issues enumerated above. If N Reactor is to be shut down in five years, or shortly thereafter, it would not be reasonable to undertake a significant and costly program of analytical modeling and experimental validation. Such a program would require three to five years to complete and thus might be completed too late to provide any benefit. Comprehensive severe accident analyses for the Savannah River reactors will be similarly costly and time-consuming. But, as discussed below, the committee believes such a program should be undertaken at Savannah River without delay.

There is a clear need in the near term to obtain a more advanced understanding of the production reactors under hypothetical severe accident conditions. A more advanced understanding of severe accidents is essential to support current efforts at both sites to extend probabilistic risk assessments from Level 1 to Level 3 studies and to develop meaningful descriptions of the risks posed by the production reactors. Reduction in risk is a critical factor in establishing priorities for upgrading the reactors. Currently a range of hardware changes, including the hydrogen mitigation system at N Reactor and changes in the confinements of the Savannah River reactors, have been proposed and are being implemented. Superior understanding of the physical and chemical phenomena that can arise in severe accidents is needed in order to ascertain if these modifications provide sufficient reductions in risk and to make sound decisions on the need for any further hardware changes in the reactors.

It is not at all clear to the committee that a sufficiently credible understanding of severe accident phenomena can be obtained in the near term by applying computer codes developed for commercial power reactors to the production reactors—the approach adopted for the N Reactor. It may be more useful to devise new, simpler models that reflect the peculiarities in the designs and the physical phenomena that can arise in the production reactors than to adapt models developed for commercial nuclear reactors. This would at least permit a limited understanding of the potential severe accident vulnerabilities of the reactors and provide the foundation for a more integrated treatment of risk, along the lines of the long-term program planned for the Savannah River reactors.

Regardless of the approach adopted for analysis of severe accidents, however, it should be recognized that the results will be subject to significant uncertainties. Substantial reductions in these uncertainties can only be achieved based on the results of programs of prototypic experimentation. This means that in the near term, in lieu of those results, large safety margins will have to be employed both in systems design and operating limits.

The committee recognizes that the rational choice of a strategy for severe accident evaluation hinges on assumptions about the remaining life of the production reactors. It is conceivable that a decision to shut down the existing production reactors, however aged they may be, could continue to be deferred by the Administration and the Congress. This raises the possibility that a decision to employ simplified models and to defer or eliminate a substantial program of severe accident research now could turn out to be shortsighted. Accordingly, decisions on reactor life and the need for alternative production capability should be confronted without delay. In this respect, the situation facing DOE at Hanford is quite different from the situation at Savannah River.

Power Excursion Accidents

One severe-accident scenario that requires prompt evaluation at Savannah River relates to the possibility of a runaway power excursion like that at Chernobyl. Severe accidents that would result in the melting of fuel and the release of radioactive material would originate from an imbalance between the power generated in the reactor fuel and the heat removal capacity of the available coolant. Such an imbalance might develop either because the ability to remove heat from the surface of the fuel has been degraded (for example, by reduction in coolant flow or loss of coolant), or because too much power has been generated in the core (such as from inadvertently withdrawing the control rods from the reactor). Because the Chernobyl accident involved the latter type of condition—a rapid insertion of positive reactivity—the committee examined the characteristics of the production reactors to assure that similar reactivity accidents could not develop in these plants. It was the development of a large, positive, reactivity feedback effect that made the RBMK reactor fundamentally unstable and was one of the major contributors to the Chernobyl accident. Once steam bubbles were formed in the coolant channels of the RBMK reactor, the interaction between higher power and more steam bubbles resulted in an increase in power, leading to a violent excursion.

As noted elsewhere in the report, the production reactors do not have positive reactivity characteristics of this type, and the potential for such a major increase in power under normal operating conditions does not appear to exist. However, in 1958 melting experiments performed on Savannah River fuel tubes in the SPERT reactor in Idaho led to the identification of a potential mechanism by which positive reactivity insertion could occur in the Savannah River reactors in a severe accident. Particles released from molten fuel might be carried upward into the moderator space in the reactor plenum after being swept from the degraded core, producing a positive reactivity effect.

Subsequent analyses in 1979 indicated that actuation of either the primary scram system or the computer-controlled backup scram system, which injects a liquid poison into the core to control reactivity, would be adequate to prevent the propagation of fuel melting for reactivity accidents initiated by the complete blockage of a Savannah River fuel assembly. However, if both systems were to fail, melting could be induced in neighboring assemblies by the combined effects of increased power and flow instability. Furthermore, calculations showed that if the blockage occurred in a target assembly of exceptionally high reactivity, even the operation of the backup system might not be adequate to prevent propagation of fuel failures and a potential runaway power excursion.

In a recent DOE technical safety appraisal of the Savannah River reactors it was recommended that analyses be made of the "potential recriticality from a molten fuel mass slumping to the tank bottom." Although the committee recognizes that the probability of such an event may be remote, it feels that more attention should be given to the potential for reactivity addition in severe accidents at both production reactor sites.

> *Recommendation: Near-term decisions on changes in the design and operation of the production reactors aimed at reducing severe accident vulnerabilities must rely on simple models and substantial safety margins. At each of the sites the Secretary of Energy should make a prompt and realistic assessment of the length of time the existing reactors are to operate. If there is a significant probability that the lives of these reactors will be extended beyond the next few years, the Department of Energy should commit to a significant program of severe accident model development and validation. Data from the program should be applied in a continuing reexamination of the risk of severe accidents and a review of the design and operation of the plants.*

> *Recommendation: DOE should ensure that the ongoing PRAs at both sites examine the risk of accidents involving reactivity addition and failure to scram, paying particular attention to*

the potential for common mode failures involving both the primary and backup scram systems or other potential anticipated-transients-without-scram (ATWS) type accidents. The design basis of the backup scram system at Savannah River should be reviewed against the results of the PRA analyses when they become available.

CONFINEMENT SYSTEMS

Conclusion: There are significant uncertainties in the abilities of production reactor confinements to mitigate radionuclide releases that would be expected to occur during severe accidents.

Production reactors differ from U.S. commercial nuclear power plants in that they use confinement systems rather than reactor containments as barriers to the release of radionuclides. A reactor containment is designed to retain both gaseous and particulate radionuclides within a nearly leak-tight volume. Release of both gaseous and particulate radionuclides would be possible, however, if the containment volume were breached during an accident. Confinement systems, by contrast, are designed to control the flow of effluents produced during reactor accidents at substantially lower pressures so that these effluents would pass from the system along prescribed pathways that are equipped to attenuate the release of the more noxious radionuclides. If the system were to operate as designed in the event of an accident, it would allow the release of some gaseous radionuclides—such as xenon, krypton, and tritium—but would prevent catastrophic confinement failure and the unattenuated release of other radionuclides.

There is no compelling evidence that mere adoption of the containment concept would substantially improve the safety of the production reactors. The committee concludes that in theory the confinement approach can be an acceptable means for mitigating accidents. The committee has some difficulty, however, with the way the confinement concept has been implemented at the production reactors. Two issues are of immediate concern:

- The ability of mitigation systems (i.e., filters and sprays) to perform adequately during severe accidents, including the capability of the confinement filters to withstand loads developed during severe accidents; and

- The incompatibility of the confinement philosophy with the discharge of liquid effluents, especially following an accident.

A summary description of the first concern is presented below. More detailed discussions are to be found in Appendix B. The issues arising from the discharge of liquid effluents from the production reactors are discussed elsewhere in this chapter and in Appendix E.

At the Savannah River and N reactors, mitigation of radionuclide releases during accidents is provided by filtration systems composed of the following components:

- Demisters,
- High-efficiency-air-particle (HEPA) filters, and
- Charcoal beds to absorb iodine.

These systems have been designed based on calculations of the release of radionuclides from degrading reactor fuel. (The calculations were not mechanistic; that is, they do not include descriptions of the actual mechanism of radionuclide release.)

Research performed in connection with commercial nuclear power plants, as well as recent studies of radionuclide release from the Savannah River reactor fuel, show that the source term assumed in the design of the confinement system may be greatly underestimated. For example, there may be greater releases of radionuclides in the form of aerosol particles and this greater mass of aerosol particulate is not accounted for in the calculations. Furthermore, the design of the confinements may not accommodate the large quantities of nonradioactive aerosol particles that might be produced in an accident. The filters used in the production reactor confinement systems are efficient at trapping radioactive particles, but they are quite limited in the total amount of particulates they can trap. If challenged in a severe accident by large quantities of particles—quantities far in excess of the filter-design capacity—the filters could overload and rupture, leading to unattenuated escape of radionuclides subsequently released from the fuel.

In addition, the pressure drop that can be withstood by the filter systems is not large—about 0.5 psig. Thus, the filters cannot be counted on to do their job unless the contractors' accident analyses demonstrate that there would be no pressure pulses sufficient to rupture the filter systems. Such analyses have not been completed. (In this connection, see the discussion of hydrogen generation and mitigation elsewhere in this chapter and Appendix G.)

Fog sprays are used in the N-Reactor confinement to augment the attenuation of radionuclide releases from the plant should there be an accident. The contractors' analyses suggest that more than 98 percent of the aerosol suspended in the confinement atmosphere after an accident would be trapped by these sprays and would not reach the filter system. These analyses assume large aerosol particle sizes (5 and 15 μm). Particles this large would be entrapped by spray droplets, whereas smaller aerosol particles would be inefficiently captured. The analyses are also based on single-compartment, contained experiments rather than on the more realistic multicompartment, "once-through" flow system of the N-Reactor confinement, with nonuniform spray coverage. The committee is aware of no demonstration that under realistic accident conditions the spray system would be as efficient as assumed in the N-Reactor safety analysis.

A fundamental assumption of the N-Reactor confinement system design philosophy is that, after an accident, release of radioactivity from the fuel occurs only after the depressurization of the reactor coolant system has been completed. Gases released to the confinement during reactor coolant system depressurization would be vented to the environment without filtration. Once depressurization had occurred, flow from the confinement would be filtered. Clearly, such a system is only effective in attenuating releases of radioactive materials to the environment if the initial depressurization is in fact completed before radioactive materials enter the system. Within the context of a Level 2 or Level 3 probabilistic risk assessment, a spectrum of severe accident sequences should be examined to determine whether there are scenarios in which accident timing could permit a significant unfiltered release to occur during initial depressurization. One such scenario might involve fuel melting while the reactor coolant system is still pressurized so that radioactive materials are released before the confinement system has been vented. Further preventive actions will be necessary if the likelihood of these scenarios is found to be substantial.

Recommendation: The Department of Energy should demonstrate whether the confinement systems at the production reactors have the capabilities (1) to withstand realistic accident *loads, and (2) to provide acceptable attenuation of realistic radionuclide releases in the event of a severe accident.*

HYDROGEN GENERATION AND MITIGATION

N Reactor

Conclusion: A concept for hydrogen mitigation has been developed for the N Reactor that involves forced mixing to prevent deflagrable hydrogen concentrations from developing, monitoring of hydrogen concentrations at key points within the confinement, and exhausting and inerting the confinement atmosphere. The committee's overall assessment of the approach is favorable. However, there are aspects of the underlying analyses and potential limitations of the proposed system that deserve careful evaluation.

The Department of Energy plans to install a hydrogen mitigation system in N Reactor by December 1987. This mitigation system will be based on several analyses currently under way. In April 1987, DOE asked the committee to review the Department's plans for providing additional hydrogen mitigation capabilities at N Reactor and to provide comments on the Department's approach to the hydrogen mitigation issue. The committee has responded to this request in a separate letter report to the Secretary of Energy that is reprinted in full in Appendix G and excerpted here:

"The protection of the public in the event of a severe accident at the N Reactor depends on maintaining the integrity of the reactor's confinement system—the safety system that is designed to attenuate the release of fission products in the event of an accident. Because the confinement has not been designed for—and is unlikely to be able to withstand—hydrogen burns of any significant size, it is important that combustible concentrations of hydrogen be prevented within the confinement, except in very localized areas near the hydrogen source.

"In order to evaluate this issue, the contractor for the facility has conducted a series of analyses in which certain defined releases of hydrogen were assumed to occur in a number of locations within the confinement. The results of these analyses indicated to the contractor that, for the assumed rate and quantity of hydrogen released, there are only two locations in the confinement where combustible limits could be achieved—the pipe-barrier space and the pressurizer-penthouse area. In other areas of the building, the contractor predicted that mixing would be sufficient to prevent a flammable concentration from being reached.

"The contractor further concluded that even if a burn were initiated in the pipe-barrier space, the increase in pressure would be dissipated in the larger volume of the confinement building and the confinement function would not be defeated. In order to mitigate the potential problem in the penthouse and to provide additional protection from the release of hydrogen in a severe accident, the contractor has developed a concept for a hydrogen mitigation system. The proposed concept is intended to satisfy four basic functional requirements:

- Provide adequate mixing between subcompartments of the building to assure that burnable concentrations of hydrogen do not accumulate.
- Monitor hydrogen concentrations at various locations within the building.
- Fill the confinement with inert gas after the initial release of steam to assure that burnable compositions are precluded in the long term.
- Provide an exhaust system to displace air to support the inerting function and to establish a lower pressure in the building than outside the building, thus preventing outleakage through the walls and bypass of the filter system.

"The first function is to prevent burnable concentrations of hydrogen. The second function provides information to plant staff that could be useful for identifying and mitigating severe accident situations. The latter two functional requirements could be just as important in that, if properly implemented, they could provide a margin of protection in the event that releases of hydrogen are greater than the assumed hydrogen source.

"The contractor considered alternative approaches to hydrogen mitigation. Preinerting and distributed ignition systems, which are methods for hydrogen control used in commercial reactors, were found not to be appropriate for the N-Reactor confinement system. The initial venting of the building atmosphere that occurs in the proper operation of the confinement system would defeat any preinerting system, and the confinement might not be able to withstand pressure pulses that could occur with a distributed ignition system, particularly for accidents involving large releases of hydrogen. This latter conclusion is consistent with the observation of pressure rises of many pounds per square inch in tests of ignition systems performed by the Electric Power Research Institute in a large-scale facility at the Nevada Test Site. Within the bounds of reasonable cost, there is no apparent preferred alternative to the concept proposed by the contractor.

“The committee's overall assessment of the general approach to hydrogen mitigation is favorable. In indicating its agreement with the general approach to hydrogen mitigation at N Reactor, the committee is *not* rendering a judgment as to the adequacy of the proposed system. Any judgment as to the adequacy of the system must be guided not only by an assessment of the general approach, but also by careful analyses of the system under various hypothetical accident scenarios, evaluation of a detailed design, and construction and operation in compliance with appropriate safety standards. Because the analysis is only at a preliminary stage, the committee's review addresses only the first of these necessary factors.

“The basic strategy of forced mixing to remove the potential for hydrogen pockets, monitoring of hydrogen at key points within the confinement, and the activation of an inerting/exhaust function appears to be sound. Indeed, the approach may provide a margin to accommodate some uncertainty relating to the amount of hydrogen generated in an accident. However, as discussed further below, there are aspects of the underlying analyses and certain potential limitations of the proposed system that deserve further careful evaluation.

“The principal aspects of the approach that merit further study or improvement may be briefly summarized as follows:

- The system design is premised on a specific accident scenario. In the committee's view, the predicted performance of the mixing and inerting systems should be examined for a broader spectrum of accident scenarios and release rates for hydrogen. For example, the assumption of the continuing operability of the Graphite and Shield Cooling System (GSCS) in the event of an accident needs a thorough examination, particularly in terms of any possible degradation of the GSCS in an accident.
- In the course of examining a broader spectrum of accident scenarios and hydrogen release rates, the contractor should extend

the mixing calculations to address localized mixing and combustion. In addition, the contractor should examine the geometric configurations near possible release points in order to assure that localized concentrations of hydrogen do not permit local detonation that could challenge the confinement or other safety systems.

- The capability of the filter system to withstand the loading of aerosols in an accident should be reviewed, and if necessary, the capability of the system should be upgraded.
- The survivability of vital sensors and mitigation equipment should be assessed. The equipment must be capable of operating in severe-accident environments characterized by wide variations in thermal-hydraulic, radioactive, and inert aerosol loads, and high radiation fields.
- The increased discharge of radioactive noble gases to the environment in the event of an accident, because of the operation of the forced exhaust system, should be examined. While operation of the proposed hydrogen mitigation system is expected to decrease the risk of failure of the confinement due to detonation or deflagration in an accident, and thus to decrease the risk of catastrophic radiation release, the exhaust of radioactive noble gases would also result in potential increases in whole-body doses to persons outside the plant. Thus, means should be investigated to assure satisfaction of the dose limits of 10 CFR 100 without relaxing the speed with which inerting takes place. Such an investigation should take into account recent research that shows that the actual quantity and chemical form of the fission products released in a severe accident would be quite different from the source terms commonly assumed in such analyses in the past."

Recommendation: The committee has recommended that its concerns be addressed in the development of a detailed design to implement the proposed approach to hydrogen mitigation. The detailed design should be subjected to an independent review before adoption.

Savannah River

Conclusion: The improved confinement system currently under development at Savannah River may cause a buildup of hydrogen in the event of a severe accident.

Plans are under development at the Savannah River reactors for an improved confinement system that would be capable of trapping noble gases and tritium during a severe accident. The committee is concerned that the overall risk of the Savannah River reactors could be inadvertently increased by some of the changes required to make the improved confinement system operate. With the existing system, the rate of airflow from the reactor room would be so high that the rate of hydrogen production during an accident would have to be very large to permit hydrogen concentrations that would reach flammable limits that could damage confinement. With the improved confinement system, however, most of the air would be recirculated and only a small flow, 2000 cfm, would be exhausted through the filter system to the environment. While this increases the ability of the confinement to retain noble gases and tritium, it also means that the system would permit hydrogen concentrations in confinement that would be substantially higher than those allowed in the current configuration.

The trade-off between mitigation of noble gas and tritium release and increasing the likelihood of a hydrogen combustion event that could disable the filter system must be weighed very carefully.

Recommendation: The proposed improved confinement system at Savannah River should be reviewed to evaluate the potential benefits and added risks of the proposed system, particularly with respect to the enhanced possibility of hydrogen-combustion events.

CERMET FUEL

Conclusion: The cermet fuel being considered by the Savannah River contractor may be susceptible to exothermic reaction under accident conditions. The contractor is currently investigating the composition and behavior of the fuel to determine whether such a reaction is precluded by reconstitution of the fuel during the manufacturing process.

Over the last several years, the Savannah River contractor has been developing a ceramic-metal composite (or cermet) fuel (U_3O_8 particles dispersed in an aluminum matrix), as an alternative to the current fuel, which consists of a uranium-aluminum alloy clad in aluminum. The committee is concerned that this type of fuel could undergo an exothermic chemical reaction during a severe accident. By a reaction analogous to the thermite reaction (a reaction between iron oxide and aluminum in a powdered mixture that is used to produce an intense heat source), significant energy release can result from the oxidation of aluminum and the corresponding reduction of U_3O_8:

$$(4/3)\mathrm{Al} + \mathrm{U_3O_8} \rightarrow 3\ \mathrm{UO_2} + (2/3)\mathrm{Al_2O_3}$$

(exoergic by about 225 calories per gram of U_3O_8 reacted). Obviously, any fuel that is susceptible to this type of reaction poses special accident risks.

The contractor is conducting analyses to determine whether chemical alteration of the U_3O_8 takes place during manufacture that will substantially reduce heat generation by the metallothermic reaction. If substantial alteration in the composition of the fuel does not occur and the reaction were to ignite locally, it is possible that the reaction could propagate. Although there is some basis for believing that the reaction would not ignite at the melting temperature of aluminum (the range of ignition temperature is 1150 to 1273 K in comparison with the aluminum melting temperature of 933 K), it is not clear whether fission-product contaminants in the irradiated fuel might reduce the ignition threshold. Furthermore, it cannot be assumed that the maximum (pre-thermite reaction) temperature that would occur in a severe accident would be limited to the melting temperature of the aluminum matrix.

The contractor is investigating the susceptibility of the fuel to ignition. In experiments with fuel to date, low-temperature initiation of the reaction has not been found. However, some evidence of a low-temperature surface reaction of aluminum with sodium uranates has been found. Continued investigations of the thermal behavior of the cermet fuel are under way.

A more complete discussion of the issues associated with use of cermet fuel is provided in Appendix F.

Recommendation: The Savannah River contractor should demonstrate that the U_3O_8 *in cermet fuels currently under development is sufficiently reduced during the manufacturing process to preclude significant metallothermic reaction in a reactor accident, or this approach should be dropped.*

HUMAN PERFORMANCE

Conclusion: Recent programs appear to have been effective in improving operator training and procedures at the production reactors. However, other activities, such as the recording and analysis of trends in human performance and the application of computers to reactor operation, appear to need further upgrading.

The accidents at Chernobyl and Three Mile Island might have been avoided but for a series of operator and management errors—errors of both commission and omission. The committee assessed operator training and performance at the production reactors in terms of five general requirements:

- Correct, clear, concise, comprehensive, and well-written operating and emergency procedures;
- Rigorous training and retraining to ensure the operating staff is qualified, including state-of-the-art training equipment, control room simulators, and procedures that develop diagnostic ability;
- Close adherence to approved procedures but clear direction of how to proceed if a procedure seems incorrect or inadequate;
- Rigorous control of all actions that could deactivate safety equipment or safety functions; and
- Correct, clear, and easily understandable instrumentation, controls, and information aids in the control room.

Operating and Emergency Procedures

On the whole, the operating procedures used at the production reactors appear to be well written and understandable, although there have been isolated instances in the last year in which certain procedures were determined to be in need of improvement or were not being followed. Such incidents, although very undesirable, do occur at all reactors. The commercial industry established the Institute of Nuclear Power Operations (INPO), in part, to define benchmarks of excellence in reactor operations, including those related to the use of operating and emergency procedures. Although the production reactor contractors have limited access to INPO, they have been successful in establishing some interaction with INPO in this area.

There are basic differences in the use of formal procedures at the two sites. At Savannah River, operators are expected to follow explicit procedures for essentially every step of the operation. At N Reactor, procedures are used for proper handling and switchover of equipment, but routine operation is based largely upon operator understanding of plant behavior. This difference is reflected in fundamentally different approaches to training at the two sites. Savannah River has developed procedurally oriented computerized diagnostic aids for its operators for both training and operation, whereas N Reactor relies more on in-depth classroom, homework, and simulator training aimed at strengthening the operator's ability to understand and diagnose plant behavior. It is not clear which of the two styles is superior; both contractors would probably benefit from analysis and incorporation of aspects of the other's approach.

With regard to emergency or abnormal event procedures and training, both sites use procedures that are based on an identification of the causes of observed deviations from normal operation. Neither site currently has an exclusive set of general procedures geared toward controlling critical safety functions in an accident. Such procedures, which are common in the commercial industry, are often called "symptom-based" procedures and are used to maintain safety conditions during an incident independent of whether or not the incident has been correctly identified. State-of-the-art symptom-based procedures at commercial reactors also provide guidance for responding to more severe accident conditions. Work is under way to develop symptom-based procedures at the production sites and to develop training appropriate to their use. Both contractors would benefit by reviewing the symptom-based approach taken by the commercial industry.

Training

Training and qualification of operators have been intensively examined at both sites since the accident at Three Mile Island. In addition, the two contractors have increased their interaction with external groups with expertise in operator training. The N-Reactor contractor has expanded contacts with INPO, the Region 4 Interoperating Utility Committee, and two programs run by the DOE Office of Nuclear Safety—the Training Coordination Programs and the Training Resource and Data Exchange (TRADE).

The Savannah River contractor has also been actively participating in TRADE and has attempted to foster closer ties to INPO, and last year sent two engineers to an INPO training course on human performance evaluation.

Training simulators that replicate the actual control rooms in the plants have recently been installed at both sites. Although they do not simulate severe accident conditions, the simulators offer a wide range of control room responses for operator training. Reprogramming of the simulators would be required to provide simulation of severe accidents, and this improvement should be undertaken.

The production reactor simulators have proven to be valuable for functions other than training. In the course of using the simulator at N Reactor, for example, the contractor recently discovered that the timing circuitry for the confinement vent valves had been improperly designed and would not have functioned appropriately in certain accident scenarios. The timers have since been rewired and the problem resolved.

Earlier in the chapter, the committee recommended that probabilistic risk assessments be completed as expeditiously as possible. Such PRAs can also play an important role in operator training to respond to severe accidents, as they have already done in the commercial industry. The PRAs can serve as models of the plants. If operators are familiar with the PRA, they will be in a better position to understand the plant's response to initiating events and possible accident progressions.

Adherence to Procedures

The committee found reasonable adherence to procedures at the production reactors, comparable to what one would expect to find at a typical commercial reactor. However, there is a need at both sites for more in-depth analysis of the underlying causes of reactor incidents, as previous reviewers have noted. Incident reports are not currently written in a way that permits a clear determination of the specific actions taken by plant personnel and how those actions might relate to what occurred. Part of the difficulty appears to lie in the understandable reluctance of incident report writers to criticize their colleagues and supervisors openly. The Savannah River contractor has tried to overcome the problem by using a reactor operations review committee as a forum for exploring incidents in greater depth, examining their causes, and identifying better means of preventing their recurrence. Clearly, unless the causes of incidents are properly identified, there is little hope of establishing meaningful trends in human performance.

Safety System Bypasses

The manipulation of safety system instrumentation appears to be strictly controlled at the production reactors. Special interlocks and detailed procedures have been developed to prevent any unauthorized bypassing of safety systems such as occurred at Chernobyl. In fact, the Savannah River contractor reviewed the Chernobyl sequence of events with operators to underscore the importance of closely following procedures when safety systems must be bypassed.

Computerized Systems

A process computer system is used at Savannah River to control routine reactor operations. It is analogous to an autopilot in an airplane in that it cannot be used during startup or shutdown and its use at normal operating power levels is optional. It is an important system at Savannah River because of the tendency toward xenon oscillations in large heavy-water moderated reactors.

One lesson from the Three Mile Island accident was the need to improve the human factors aspects—the interface between operators and equipment—of nuclear reactor control rooms. In response to this need, the Savannah River contractor has developed a computerized diagnostic aid known as the Diagnosis of Multiple Alarms (DMA). The DMA system is not a type of Safety Parameter Display System (SPDS) like those installed in commercial reactors after the Three Mile Island accident. Rather, it is a computerized system that employs a specific set of truth tables to identify and prioritize the operating procedures that should be used by the operators in response to deviations from normal operation.

The purpose of the DMA system is to perform in a short time the analysis that operators would otherwise have to perform during an emergency, at a time when numerous alarms have been activated in the control room. In such a situation, the operators might have too much information to select the proper response.

The DMA system selects the appropriate procedures and lists the order in which they should be implemented. As with the simulator at N Reactor, development of the DMA system has led to unanticipated results. A number of errors in logic in the existing Savannah River procedures were discovered during the development of the DMA system.

Implementation of suggestions from the post-Three Mile Island reviews also resulted in improvements to the N-Reactor control room. Unused switches were removed from control panels and the control room alarm was replaced. The committee has been advised that a more recent review of the control room has been conducted and that an evaluation is in progress to determine whether additional changes are needed.

The current trend in production reactor operations is toward greater control of plant equipment using digital systems. This appears to be occurring, however, with very little oversight from DOE. It is important to recognize that while computerized systems may reduce the likelihood of certain kinds of human error, they do not eliminate error; computerization merely shifts the burden of performance from one human activity to those specifically related to the use of computers, such as software engineering. The Savannah River reactors, for instance, have computer control systems governing the operation of their "charge-discharge" machines (which are used to remotely insert and remove fuel and target assemblies from the reactors). Although the machines have been operated successfully through many charge-discharge operations, there have been a number of recent incidents involving breakdowns of one kind or another. The contractor plans to replace the existing charge-discharge system computer with a newer, larger memory system, but the kinds of incidents that have occurred reflect the more general need for better software reliability.

It should be noted that the recent DOE Design Review recommended replacing the contractor's software quality assurance programs with a more rigorous, mandatory program. Such a program has been initiated by the contractor in the current fiscal year. The program elements include code configuration control, documentation requirements, coding practices, validation and verification, and assignment of responsibilities. At the planned level of effort, it will take the contractor four years to work through the current backlog of codes. In the same report DOE also recommended maintaining on a continuous basis a coterie of qualified personnel who could serve as backup to contractor staff qualified to work on the reactor control and safety computer software. Since this second program is run by the plant and the program discussed above is run by the laboratory, the Department should ensure that both plant and laboratory software reliability programs meet the same high standards.

N Reactor has experienced similar problematic behavior from its computerized systems, and in particular the controls governing the plant's secondary system (the byproduct steam system). Causes of these problems include misalignment of components within the control machinery and poor software. In response, the contractor has recently adopted stricter rules governing software changes.

Very high reliability is required for the software that is used in the design of the core loading for each cycle at Savannah River. Since each core composition is different, a detailed reactor physics analysis must be performed prior to loading the fuel and target assemblies. Thus, many opportunities are presented for errors to occur which could lead to severe consequences, such as inadvertent criticality during loading or melting of an assembly. The committee believes that the adequacy of the existing quality assurance procedures to control the frequency of core configuration design errors should be examined from a probabilistic viewpoint in the ongoing PRA.

> *Recommendation: The DOE contractors should expedite the development and incorporation of procedures and training aimed at restoring critical safety functions and controlling critical safety parameters in the event of abnormal conditions.*
>
> *Recommendation: More thorough analyses of the causes of reactor incidents should be conducted in order to improve human performance.*
>
> *Recommendation: Better methods for analyzing trends in human performance should be developed.*
>
> *Recommendation: The management of software engineering activities should be reviewed and improved. DOE should sponsor an independent review of computerized systems at both sites to ensure that the systems are fault-tolerant.*

LIQUID EFFLUENTS

Conclusion: Discharge of radionuclide-contaminated liquids from the production reactor confinements into open basins, as occurs during normal operation of the N Reactor and as could occur during accidents at both sites, poses a safety hazard. It is also an environmentally unsound practice.

In the event of a severe accident at the Savannah River reactors or the N Reactor, radionuclide-contaminated liquid effluents would be discharged from the plant. At Savannah River, the discharge would be first to a 60,000-gallon tank and then to a 500,000-gallon tank, both of which are vented back into the reactor confinement. Overflow from the tanks, however, would be to an open, 50,000,000-gallon basin. Flow to the 500,000-gallon tank in an accident could be as high as 14,000 gallons per minute, so overflow into the open basin could occur in as little as one hour.

Discharge of liquid effluents at N Reactor both under normal operating conditions and during an accident is to a 6,300,000-gallon unlined pit or "crib." The liquids drain into the ground below the pit allowing the possibility of eventual contamination of the groundwater and the introduction of elevated levels of radionuclides into the Columbia River. Contamination of the liquid effluents from N Reactor during an accident would be especially severe since substantial attenuation of airborne radionuclides from the reactor is achieved with water sprays and the contaminated spray water would be discharged to the "crib." Congress is currently considering a line item in the FY 1988 defense authorization bill that would provide monies to build a Waste Effluent Treatment Facility for the N Reactor.

In the past, discharges to the "crib" during normal operation of the N Reactor have averaged about 1330 gallons per minute. These discharges amounted to releases of about 6000 curies per year of radioactivity with half-lives greater than 48 hours—primarily radioactive isotopes of cobalt and strontium. Discharge of these contaminated liquids from the N Reactor is clearly an environmentally unsound practice and has been criticized as such in previous safety reviews. Indeed, only a waiver of a currently effective Department Order has allowed the practice to continue. During the current standdown of the reactor, the contractor has made modifications that are expected to reduce the discharges during normal operation to about 505 gallons per minute and about 4100 curies per year. The contractor is now working on a filtration system that would further reduce the radioactivity of the discharges to about 1900 curies per year and has proposed installation of an ion exchange system, which has not yet been funded, that would reduce the discharge to about 50 curies per year.

Under accident conditions, the discharge of radionuclide-contaminated liquids from the production reactors violates the confinement concept of defense-in-depth against radionuclide release and constitutes a safety hazard. The hazard under accident conditions is not merely eventual contamination of groundwater. Rather, it is that radionuclides—especially iodine—can vaporize from the discharged liquids. Those radionuclides that vaporize from the liquid effluents would be released to the environment and would thus have bypassed the protection that is meant to be provided by the confinement system.

Consideration of iodine vaporization from water pools has been included for some time in the analysis of reactor accidents at commercial nuclear power plants. (A detailed technical discussion of the chemical processes involved in iodine vaporization is provided in Appendix E.) Vaporization can occur because molecular iodine can be formed in water by oxidation of iodide ions:

$$2\,I^- + 2\,H^+ + (1/2)O_2 \rightarrow I_2\ (\text{aqueous}) + H_2O.$$

Though some hydrolysis of the iodide to HOI occurs, further oxidation to iodate, IO_3^-, is kinetically slow. Consequently, the dissolved molecular iodine can vaporize:

$$I_2\ (\text{aqueous}) \rightarrow I_2\ (\text{gas})$$

and the vapors can be carried away by ambient air currents. The extent to which molecular iodine will be formed depends on the water temperature, hydrogen ion concentration, total iodine concentration in the liquid, and time. There is evidence that the radiation field produced by the dissolved radionuclides can enhance the vaporization of iodine from water pools. Thus, accidents at the production reactors could lead to significant airborne releases of radioactive iodine, yet these releases have apparently not been properly considered in accident analyses for the plants.

In addition, iodine is quite reactive, especially in a radiation field. In particular, iodine in aqueous media will react with many organic materials, including plastics, to form CH_3I, which will vaporize from the water. Plans to upgrade basins at the N Reactor by lining and covering them with polyethylene sheets may be frustrated by the reactive nature of iodine in intense radiation environments.

Recommendation: The Department of Energy should establish ways to protect the environment from the radionuclide-contaminated liquid effluents discharged during normal operation of the N Reactor. Means should also be found at both Hanford and Savannah River to prevent the direct discharge of contaminated liquid effluents to the environment following an accident.

EMERGENCY PLANNING

Conclusion: Emergency planning and response capabilities at the Department of Energy's production reactor sites appear to be comparable in terms of onsite preparedness to those at commercial reactor facilities. Offsite planning and preparedness, however, do not appear to be on a par with those of localities surrounding a commercial reactor.

The nuclear accident at Chernobyl involved the prolonged and extensive dispersion of large quantities of radioactive materials throughout the environment. Aspects of the response by Soviet authorities to the accident appear to have been unprecedented. To date, the measures taken by the Soviets to respond to the accident have not been thoroughly assessed in order to determine whether there are any lessons that can be learned that would improve emergency preparedness in the United States.

There are aspects of emergency preparedness at Hanford and Savannah River that are significantly stronger than at most commercial U.S. nuclear power plants. First, the size of the sites and low population density around them significantly ease the difficulty of providing effective planning and response. Although there are large numbers of onsite workers at each site, they are generally clustered in a few locations and should be relatively easy to protect in the event of an emergency. Second, the large numbers of facilities and contractor organizations located at the sites make available extensive resources (for example, knowledgeable staff and equipment for radiation monitoring and meteorological assessment) that could be marshalled in the event of an accident. Both sites also indirectly benefit from proximity to commercial nuclear reactor facilities that have their own emergency preparedness activities, including substantial involvement of state and local government organizations. The Vogtle Nuclear Plant is located about 10 miles from the Savannah River site, and the Washington Nuclear Plant 2 is located within the Hanford site boundary. Although the committee did not explicitly examine the relationship between DOE's field offices and state and local government response organizations, DOE should assure that the emergency preparedness of those organizations, and their communications with them, are adequate to protect the health and safety of offsite individuals.

DOE has conducted audits of the emergency planning organizations and procedures at Hanford and Savannah River over the past six years. These have resulted in a series of recommendations for improvements to facilities, procedures, and training. The most recent appraisal for the Savannah River production reactors concluded that the emergency preparedness program was effective and provided three recommendations for improvements. One recommendation was that the Savannah River contractor revise its classification of emergency response levels so they conform with DOE and commercial nuclear industry practice. This recommendation had been made previously, in reviews following the Three Mile Island accident, but was not implemented by the contractor. The committee understands that actions have now been implemented in response to each of the three items identified by the appraisal team.

DOE's most recent appraisal of emergency preparedness at N Reactor also concluded that the contractor's emergency response organization and facilities were generally adequate to protect the health and safety of the public in the event of a reactor accident. The appraisal included several recommendations for improvements, and actions have either been implemented already or are planned to address each of the specific items cited by the appraisal team.

The purpose of emergency preparedness is to provide persons responsible for action in the event of an emergency with adequate knowledge, understanding, equipment, and facilities for making and implementing reasoned decisions. It is important that the planning basis consider the entire spectrum of events that might occur, from benign to severe. DOE Orders (5500.2, 5500.3) require that all "credible" accidents be analyzed and that estimates of the characteristics of associated accident scenarios and source terms be used as a basis for emergency planning.

Within the limits of the existing definition and analyses of credible accidents at the production reactors, the Hanford and Savannah River emergency planning organizations, staff, facilities, and training appear to be generally quite good. Frequent exercises of the response staff and procedures have been held at each site, and Savannah River emergency planning staff participate in Federal Emergency Management Agency (FEMA) evaluations of commercial reactors and in federal-level radiological emergency exercises. However, DOE and its contractors do not fund or participate in annual preparedness activities with state and local governments. The ability of emergency response organizations to cope with accidents beyond the current design basis (for example, full core melt accidents with confinement and filter failures) needs to be more fully addressed. As results and insights are gained from ongoing and future severe accident analyses and PRAs, they should be reflected in changes to existing emergency plans and in future exercises. Finally, the committee believes DOE should assure itself that both contractors have adequate answers to such questions as: (1) Given a slow-developing accident, what criteria should be used on which to base a decision to evacuate and what information is needed to assess the criteria? (2) Given an accident in progress with releases to the environment having already occurred and still under way, how best can the extent, distribution, and movement of the radioactivity in the outside environment be monitored so as to make rational decisions regarding emergency actions? (3) Is the information gathering network on traffic conditions, weather, and competing emergencies adequate for on-line, ad hoc decisionmaking? (4) Is there a readily accessible data base available on the local hydrology, transportation routes, topography, demography, temporal changes in the population density on an hourly basis, and meteorology? and (5) Is there a site-specific atmospheric transport model available at the emergency response center that can be used interactively with real-time weather data to project the movement of the radioactivity for short times into the future in order to predict who is at risk and when?

Recommendation: Emergency planning activities at the Department of Energy's production reactor sites should consider the entire spectrum of accidents that could occur, going beyond design-basis accidents to include large-scale core melt accidents. As results and insights are gained from ongoing severe accident analyses and probabilistic risk assessments, they should be reflected in changes to emergency plans, training, and facilities.

The U.S. Department of Energy and its contractors should fund and participate in annual preparedness exercises with state and local governments.

The Department of Energy should also analyze those lessons learned from the Chernobyl accident that relate to emergency planning and response to determine if improvements in regional and national capabilities are warranted.

3

Strengthening the Technical Basis of Reactor Safety Management

THE ROLE OF THE DEPARTMENT OF ENERGY IN HISTORICAL CONTEXT

The Department of Energy and its predecessor agencies have been responsible for the production of defense nuclear materials since that task was originally assigned to the Atomic Energy Commission (AEC) in 1946. In 1974 the AEC was reorganized into an independent Nuclear Regulatory Commission (NRC) and a program-oriented agency within the Executive Branch (the Energy Research and Development Administration or "ERDA"); the latter became the basis for the Department of Energy in 1977.

The AEC led the way in reactor safety. It was an agency formed at the beginning of the atomic age with the sole mission of developing and controlling the use of nuclear technology. The DOE, on the other hand, is a department of the Executive Branch with many nonnuclear responsibilities. Despite its origins in the AEC, DOE's potential for leadership in nuclear safety technology has been constrained by the assignment of regulatory functions to other organizations and by the preeminence within the federal government since the mid-1970s of the view that nuclear technology ought to be developed principally by private industry.

Commercial experience with nuclear reactor operation has now become far more extensive than that of the DOE. The commercial nuclear industry in this country currently encompasses more than 100 nuclear power plants. It is regulated by the Nuclear Regulatory Commission, which has no responsibility for DOE-owned reactors. For the most part, national leadership on matters of nuclear safety has shifted to the NRC in its role as regulator of commercial nuclear power plants.

Funding constraints have also acted to diminish the role that DOE has been able to play in the area of reactor safety. For many reasons—including the many demands that were placed on the DOE in the late 1970s in the wake of rapid increases in international oil prices—DOE reactors did not receive the same funding priority as they had in prior years. As a result, reactor upgrades and even some plant maintenance were deferred. Mounting budgetary pressures in the 1980s restricted production reactor funding so that the DOE safety program was devoted largely to catching up with deferred maintenance. Upgrades intended to enhance the safety of the plants received limited funding or were postponed. As noted in Chapter 2, these factors have had lingering effects. The effects can be seen today in such practices as the treatment of liquid effluents from the production reactors and the N Reactor's antiquated methods of refueling.

The current DOE mode of operation differs from the mode of operation that existed during the days of the AEC. At both agencies the basic strategy was to delegate maximum responsibility to the contractors, and in each case most of the technical expertise has resided there. Yet during the AEC era, there was also a small group of the highest technical quality in the headquarters staff whose role, simply expressed, was to keep the contractors' feet to the fire. This small headquarters staff usually had adequate influence, because its members were as technically competent as those in the field, and it was widely understood that they had the full backing of the headquarters program directors. The AEC also had good reason to assume highest quality in the contractors. The people who had designed the reactors were contractor employees. Until recently, some of them were still on board. They knew every nut and bolt in the reactors they were operating. In sum, with the breakup of the AEC and the passage of time, the technical experience in headquarters, in the field, and in the contractors' staffs has diminished.

The breakup of the AEC also led to a loss of outside oversight of the production reactors. NRC plays no role in assessing the safety of DOE reactors, and there is no standing committee of non-DOE advisors paralleling the Advisory Committee on Reactor Safeguards (ACRS). The ACRS, which continues to provide advice on reactor issues to the NRC, was originally created in 1946 to assist the AEC. At the outset, the ACRS reviewed only AEC production, research, and test reactors since there were as yet no commercial reactors.

One of the first issues addressed by the ACRS concerned the suitability of the Hanford site for the production of plutonium. For years the ACRS maintained separate subcommittees on the Hanford and Savannah River production reactors, which met frequently and provided a steady flow of expert and independent advice. With the abolition of the AEC, regular review of the production reactors by the ACRS ceased, and the ACRS subcommittees on the production reactors were disbanded. Although the system was not perfect—for instance, ACRS had no mechanism for ensuring that its concerns were resolved—the ACRS did focus attention on important issues and encourage improvements in the safety of the production reactors. The Secretary of Energy is authorized, however, by 42 U.S.C. 5814 to seek advice from the ACRS. Although DOE has used this authority to acquire ACRS review of new reactor designs such as the Fast Flux Test Facility, it has not sought ACRS advice on the production reactors, and no other body has been created to assume ACRS's role.

The accident at Three Mile Island (TMI) in 1979 had a major impact on the regulation of commercial nuclear power. As a result of the accident, the NRC and commercial nuclear utilities embarked on a wide-ranging and expensive program to improve commercial reactor safety; the implementation of this program is still in progress today. Although DOE played a part in the process of extracting lessons from the accident, it has been slow to incorporate those lessons in the operation of its own reactors. In mid-1980, the Under Secretary of Energy formed an in-house panel to study the safety of DOE's large reactors in light of the TMI accident. The study, however, was narrowly focused on operator qualifications and training. Even so, the panel's report clearly pointed out, among many other findings, that DOE was not effectively carrying out its responsibilities with respect to nuclear safety, and furthermore, that it did not have the in-house capability to do so.

Six years later, only one of the fundamental recommendations made as a result of the DOE's study has been implemented—elevation of the functions of environmental protection, safety, and health to the level of an Assistant Secretary. The equally important recommendation to establish effective outside oversight of the DOE reactors still has not been addressed. Recently, however, the Department has indicated a willingness to have some form of external oversight established.

Following the Chernobyl accident, DOE moved to reevaluate its reactors. In addition, between November 1986 and November 1987 DOE took a number of highly visible actions at three DOE reactor sites:

- DOE extended the N Reactor's annual outage in order to implement a substantial portion of the many safety enhancements recommended by internal and external reviewers who examined the plant in the wake of the Chernobyl accident.
- At Savannah River, DOE reduced the reactor power level by 50 percent and required the contractor to develop an acting plan designed to assure that adequate emergency core cooling capability could be demonstrated.
- DOE shut down the High Flux Isotope Reactor for safety reasons and placed other facilities at Oak Ridge on standby because of poor management practices.

These actions responded to findings of specific deficiencies. As dramatic as these measures have been, they have not answered the underlying question of whether DOE can fully discharge its safety responsibilities with the resources and mangement practices it now uses.

THE CURRENT MANAGEMENT SYSTEM

The production reactors are operated by contractors selected by DOE. At both Hanford and Savannah River the contractors maintain large staffs that are responsible for reactor operation, engineering, safety, and environmental protection, among many other functions. The contractors benefit from staffs that are experienced and reasonably technically sophisticated and are led by senior managers who typically have long-standing familiarity with the reactors. Each site also has the benefit of an associated support laboratory of national repute. As one reviewer noted in 1985, "the DOE's fundamental operational tenet is to put responsibility for safety primarily upon the contractors." Although this delegation of operational responsibility to contractors may be appropriate, the assignment does not relieve DOE of its legal mandates to assure public safety and to supervise contractor performance.

Within DOE the primary responsibility for the safe operation of the reactors lies with its line management. Authority flows from the Secretary, through the Under Secretary, to the managers of the local operations offices at the two sites. The managers of the operations offices at the Hanford and the Savannah River sites are the DOE officials who are primarily responsible for providing program direction to the contractors, and who must assure the implementation of adequate environment, safety, and health programs. By comparison with the contractor organizations, the staffs of the DOE operations offices are relatively small. As currently structured, both the Savannah River and Richland operations offices each have one division responsible for the actual operation of the reactors and a separate division, reporting to the manager, responsible for auditing the performance of the contractor in terms of compliance with safety, quality assurance, and environmental requirements.

Although the manager of the operations office plays a key role in assuring the attainment of production objectives and the safe operation of the reactors, the manager's activities are significantly affected by at least two other DOE officials. The production goal and the budget allocation for the reactors is developed by the Assistant Secretary for Defense Programs (DP). Control over the budget gives this Assistant Secretary significant power to affect the activities of the operations office and the contractor. (The manager of the operations office, nevertheless, may independently raise budget issues with the Under Secretary as may the Assistant Secretary for Environment, Safety, and Health.) In addition, proposed plant modifications, changes in technical specifications, and safety analyses must be approved by DP.

Within the DOE, oversight of the safety, quality assurance, and environmental performance of both the DOE line management and the contractors is the responsibility of the Assistant Secretary of Environment, Safety, and Health (ES&H). ES&H conducts intensive appraisals of the production reactors and the contractors' activities. This oversight function has been significantly upgraded in the past year and a half. The size and technical depth of the permanent staff of ES&H is still not large, especially in relation to the size of the defense production enterprise as a whole, although the ES&H staff is augmented for the purpose of conducting site appraisals by the use of consultants.

In sum, the operations offices are responsible to two separate offices in the DOE headquarters. The effectiveness of this arrangement depends on the balance of capabilities in those two offices and their relative authority.

PRINCIPAL AREAS FOR IMPROVEMENT

Under the current set of arrangements, ES&H is charged with coordinating with DP in the establishment of departmental safety standards and with independently monitoring the safety performance of the contractors and the operations offices. Both ES&H and DP have access to the Under Secretary and the Secretary, so that in principle these officials may also become directly involved in any given decision. Yet, in the absence of independent external review, both of these headquarters organizations and the Under Secretary depend on the field offices and the contractors for information about the day-to-day operation and level of safety achieved in the plants. Except on an ad hoc basis, the Secretary has no access to objective external expertise, which would provide independent judgment of the quality of the standards set by ES&H, the supervision provided by the field offices, or the performance of the contractors.

The committee's conclusions and recommendations as to DOE's management of the reactors relate to three principal findings: (1) DOE's overreliance on its operating contractors; (2) weaknesses in the current implementation of the DOE management approach; and (3) the need for increased outside involvement.

DOE's Relationship with Its Contractors

Conclusion: The Department of Energy, both at headquarters and at the Richland and Savannah River operations offices, has relied almost entirely on its contractors to identify safety concerns and to recommend appropriate actions. In large part this results from a marked imbalance in technical capabilities and experience between the contractors and the DOE staff.

The contractors responsible for the operation of the N Reactor and the reactors at Savannah River have excellent records of safe operation. There have been no major reactor accidents at these facilities. Both facilities have records of avoidance of lost workdays as a result of on-the-job injuries at least 10 times better than that of U.S. industry as a whole.

Perhaps because of the contractors' records of accomplishment, DOE has tended to defer almost exclusively to the contractors' expertise and to rely on the contractors to identify, evaluate, and resolve safety issues. This appears to be due partly to the imbalance in levels of staffing and partly to differences in technical training and experience, as discussed below.

The contractors at the production reactors have a large permanent staff, while the DOE presence on these sites is relatively small. For example, the DOE field organization staff charged with oversight of operations at N Reactor consists of only eight to ten professionals, the auditing group includes only one person with full-time responsibility for the reactor, and the safety group has two to three professionals with full-time responsibility for the reactor. This contrasts with a contractor staff of approximately 1200 at N Reactor. One DOE official explained that the Department's approach is to "skim over the surface" in the hope of "sensing" problems that justify closer examination.

The DOE staff also does not have a depth of technical experience commensurate with that of the contractor staff. (At Savannah River the greater experience of the Du Pont staff is diluted to some extent by Du Pont's traditional practice of continually rotating mid-level staff in and out of positions of responsibility.) At Richland, for example, the committee was told that the typical DOE staff member entered employment with DOE as a first job upon graduation from college and has been in DOE's employ, on average, for two to three years. These persons obviously possess no reservoir of technical experience and have not benefited from extensive contact with the external technical community. Moreover, DOE technical staff may leave government service after a short tenure, or move on to other staff or management roles.

The inherent limitations of this approach are demonstrated by the events that led to the current 50 percent reduction in power of the Savannah River reactors. As noted in Chapter 2, in order to establish allowable upper limits for reactor power, the Savannah River Laboratory conducted experiments in 1979 focusing on the capability of the Savannah River reactor emergency core cooling systems. By 1981 personnel in the contractor's reactor department had become convinced that the 1979 experiments were deficient. Thus, uncertainty existed within the contractor organization concerning the validity of the existing limits of safe operation, as defined by the design bases of the emergency core cooling systems. Nevertheless, the reactors were operated at full power for the next five years while the laboratory devoted relatively few resources to validating the 1979 experiments. At no time during the entire five-year period did anyone at DOE become aware of the uncertainties that existed regarding the power limits then in use. To its credit, DOE acted within hours after being informed in November 1986.

The committee views this case as indicative of the underlying nature of the DOE-contractor relationship; it is unusual only in the severity of the corrective action ultimately taken. The committee concludes that DOE places undue reliance on its contractors for the assurance of safe operations. Safe operation of the production reactors requires independent and competent assessment of contractor decisions by experts in reactor safety who are at arms length from day-to-day operations. While DOE oversight must allow a contractor a degree of technical discretion and must demand individual accountability, DOE involvement has to be comprehensive enough to allow it to evaluate all aspects of the contractor's performance. The DOE will not be capable of recognizing and rewarding good practice or recognizing and forestalling improper action unless it knows and fully understands what the contractor is doing.

> *Recommendation: The Department of Energy should acquire and properly assign the resources and talent necessary to ensure that safe operation is being attained. Changes in staffing levels and budgets will likely be required to achieve such a capability. The purpose of these changes should be to provide the staffing and funding necessary to ensure comprehensive involvement by DOE in the safety assessment and operation of the production reactors. In addition to establishing comprehensive involvement with the contractors, DOE should clarify and strengthen its reporting requirements to ensure that the contractors immediately bring matters of safety consequence to the Department's attention.*

Management Structure

Conclusion: The Department of Energy's management approach copes with the mix of production and safety responsibilities faced by the Department, but falls short of reasonable expectation.

Although the independent ES&H organization created in the last year has conducted several detailed appraisals of reactor operations, areas of weakness remain:

- ES&H has no permanent onsite presence. The ties between DP and the operations offices are relatively strong, whereas the ties between ES&H and the field offices are weak.
- The capability of the operations offices to conduct ongoing surveillance programs and to review proposed projects needs to be strengthened.
- Although the expanded ES&H organization in DOE headquarters has sponsored important audits of the production reactors over the last year, the audits have been episodic and narrowly focused. They do not substitute for continued onsite surveillance or in-depth review of contractor safety analyses.
- The DOE has nothing fully comparable to the offices and divisions of the NRC charged with research, reactor regulation, inspection, and event analysis. The functions performed by these offices and divisions are not effectively performed by the DOE.
- The liaison of DOE with the external technical community falls short of what is desirable and easily possible.

The committee recognizes that the DOE-contractor relationship differs qualitatively from that which exists between the NRC and its licensees. The balance between costs and benefits is achieved in the context of commercial plants through the relationship between regulator and licensee and with the involvement on economic matters of public utility commissions. DOE, in contrast to the NRC, must play a mixed role. It must assume the position of the regulator in terms of assuring the safety of operations, but it must also assume the role of licensee through the scheduling and financing of operations to meet program requirements. In this scheme, the contractors act as third parties with interests in both safety and production.

The tension between DOE's two roles is apparent at the production reactors, particularly in connection with the implementation of safety modifications. As noted earlier, the committee was told in interviews that, in the past, the contractors have pressed for safety upgrades that DOE has then rejected for budgetary reasons.

DOE judgments about plant upgrades do not appear to reflect the results of explicit cost-benefit analyses; indeed, so far as the committee could determine, there is no formal system currently in use to assess proposed production reactor modifications that takes public health risks explicitly into account. As previously noted, where cost-benefit analyses of plant upgrades have been proposed, reduction of avoidance of public health risks have not appeared explicitly in the computations. The committee believes that the lack of a meaningful procedure for conducting cost-benefit analyses of proposed plant modifications and the lack of a clear-cut, risk-based safety objective are two sides of the same coin.

In light of the conflicting responsibilities of the Department to meet production requirements and assure safety, the committee considered whether to recommend the creation of an entirely new management structure responsible solely for assuring the safety of the production reactors. For example, the safety-oversight responsibilities now allocated to DOE might be transferred to the NRC or to an analogous new entity created outside DOE. But either of these options might lead to interagency conflicts and disruption that could inhibit both safety and production. The committee concluded that DOE can accomplish the reactor safety functions assigned to it by Congress *if it dedicates itself to the task*. Thus, the committee has recommended measures to strengthen and improve the Department's safety oversight by modification and strengthening of the existing management structure. The committee acknowledges, however, that the passage of time may demonstrate that more radical measures should be adopted.

In the committee's view, significant modifications of the Department's approach are necessary. The committee considers it essential for the DOE to strengthen the internal monitoring capability it has established for the production reactors. The monitoring capability itself should be free of the demands presented by program responsibilities. DOE has recognized this need in its efforts to build an effective ES&H organization. The committee takes note of and commends the efforts of that office in the review of design and operational issues at the production reactors, but urges further improvement.

Recommendations: In strengthening the monitoring of safety at the DOE production reactors:

- *The Department of Energy should expand its capability to sponsor research, conduct and review safety analyses, evaluate operations, analyze trends, and assess proposed plant upgrades.*
- *ES&H should have a permanent and significant onsite presence with a formal reporting relationship between onsite personnel and headquarters staff.*
- *ES&H should have more direct access to and involvement in the resolution of key safety issues on a timely and effective basis.*
- *ES&H should be centrally involved in the Department's internal budgetary processes so as to help assure adequate funding for safety improvements.*

External Oversight

Conclusion: The Department of Energy's safety oversight of the production reactors is ingrown and largely outside the scrutiny of the public. Weaknesses in management of the defense production reactors have led to a loose-knit system of largely self-regulated contractors operating within budgetary constraints imposed by and on the Department of Energy.

The DOE is not receiving external and independent review of its reactor safety decisions of the type regularly obtained in the commercial reactor industry. There is no organization within or associated with DOE that exercises reactor safety responsibilities analogous to those of the ACRS. There is no organization within or associated with DOE that can review reactor operations in a manner similar to that exercised by the Institute for Nuclear Power Operations (INPO) for commercial reactors. DOE has also made little use of the services of independent safety review committees such as those employed routinely by individual utilities to examine safety issues at commercial plants. As a result, DOE and its contractors do not regularly benefit from the knowledge available elsewhere in the nuclear community.

The concept of external, independent, or peer review is not foreign to the Department of Energy. Parts of the Department use review committees, such as the Energy Research Advisory Board (ERAB), which provides input to the Director of the Office of Energy Research. The DOE national laboratories (such as Argonne National Laboratory and Brookhaven National Laboratory) have outside peer review in the form of visiting committees. These committees are usually composed of scientists and engineers from universities, industry, and other national laboratories. They report to the trustees of the universities that administer the laboratories for DOE, providing copies of their reports to laboratory management and to local DOE operations offices.

At the DOE production reactors, the committee found little in the way of external review. The relationship between the production reactor contractors and INPO has been limited and of limited utility to DOE and its contractors. At Savannah River, several small external review groups have been formed for narrow technical purposes, and there is a Reactor Safety Advisory Committee. However, the latter meets infrequently, does not set its own agenda, does not have staff support, and has few members unaffiliated with DOE or the contractors.

Independent review has been of enormous importance at commercial reactors, providing new and workable views and insights and assuring the incorporation in safety decisions of operating experience at similar facilities throughout the world. Such a review is virtually nonexistent at DOE production reactors. Accordingly, the committee concludes that there is a critical need for additional external oversight at both Savannah River and Hanford.

Moreover, an independent and aggressive oversight committee would help in building public confidence that these reactors are indeed being operated to high safety standards.

Recommendations: An independent external safety oversight committee, advisory to the Secretary of Energy, should be established. The oversight committee should possess the following features:

- *Members should be of recognized stature with expertise covering the full range of disciplines relevant to reactor safety.*
- *Members should include individuals from outside the DOE community.*
- *The committee should have authority to set its own agenda.*

- *The committee should be authorized to review both the product and the process of the Department of Energy and contractor efforts, including review of design, safety analysis, operations, management, inspection, and enforcement.*
- *The committee should be supported by a full-time, technically qualified staff, without which it cannot function effectively, and by a budget adequate to obtain external technical assistance as required.*
- *The bulk of the committee's work should be unclassified and available to the public.*

The Department of Energy should also encourage each contractor to establish a permanent visiting committee of outside experts to review and assess the implementation of safety initiatives and to report to contractor management at the site and at corporate headquarters. These committees should be free to set their own agendas.

Follow-Up

The committee believes that the foregoing recommendations, if implemented, could help foster a self-sustaining system of safe operation and oversight: operating units working to well-formulated safety standards, a strong internal monitoring and enforcement capability, and external oversight to advise on the adequacy of safety goals and DOE's adherence to them. Given the difficulty of implementing organizational changes of the scope discussed in this chapter, however, the committee believes that even if such a system were adopted there would be a need for evaluation after a reasonable period to see if it is functioning as intended.

It is worthwhile pointing out that the committee's recommendations do not lead to simple, unequivocal answers to questions related to the appropriate regulatory model for achieving production reactor safety. Many of the issues discussed here are ones that need to be addressed by measures other than regulation: safety goals are instrumental to the development of a cost-effective safety system; the balance of technical capability between DOE and its contractors needs to be restored to bolster the DOE role and improve the DOE-contractor relationship; the transition between the existing set of aging facilities to a modernized production system must be managed in a tight budget environment. None of these imperatives can be met through regulation per se.

Finally, the technological vigilance required to assure safety at the DOE reactors cannot be generated from organizational structure alone. Within even a large, properly structured organization, safety is a reflection of institutional commitment and capability. Leadership at the policy-making level is essential, and dedication to safety must permeate the Department of Energy. The DOE has made recent strides in this direction, but those efforts need to be bolstered, institutionalized, and sustained.

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Appendix A
Statement of Task

The National Research Council will undertake an assessment of safety and technical issues raised by the nuclear reactor accident at Chernobyl. The assessment will focus on the eleven Class A (over 20 MWT) reactors operated by the Department of Energy. These eleven reactors are the N Reactor at Hanford; the C, K, L, and P production reactors at Savannah River; the Fast Flux Test Facility at Hanford; the Experimental Breeder II and the Advanced Test Reactor at Idaho Falls; the High Flux Beam Reactor at Brookhaven; and the High Flux Isotope Reactor and Oak Ridge Research Reactor at Oak Ridge.

The committee of individuals expert in nuclear reactor safety, risk analysis and assessment, and management of large production and research programs, will carry out the review. The committee will obtain the results of DOE's ongoing safety assessments of production reactors and will receive briefings on what is known about the Chernobyl accident. In addition, the committee will respond to Secretary Herrington's request for public participation by holding public meetings to receive ideas and information from interested individuals and groups.

This Statement of Task is taken largely from the August 1986 contract between the U.S. Department of Energy and the National Academy of Sciences-National Research Council.

The committee will report within nine months on matters of immediate safety concern for DOE reactor design, construction, and operation. It is expected that the committee will consider a number of more generic, perhaps longer-term, safety issues that arise from a more complete understanding of the Chernobyl accident; these will be discussed in a later report.

Appendix B
Confinements: Technical Discussion

"Defense-in-depth" has been the safety philosophy adopted in the United States for nuclear reactors [1]. Implementation of this philosophy in large nuclear reactors has involved erecting multiple barriers to the release of radionuclides into the environment. The cladding on reactor fuel and the reactor coolant system itself are the first two barriers, while the final and most robust barrier in commercial nuclear power plants is the reactor containment building. In the production reactors at Savannah River and at Hanford, the final barrier is the reactor confinement.

Both reactor containments and reactor confinements are intended to function as barriers to radionuclide release in the event of a very damaging reactor accident that breaches all other barriers. These two concepts function, however, in different ways. Reactor containments are designed to retain all radionuclides released from the fuel—radionuclides in both gaseous and particulate form—in a nearly leak-tight volume. Of course it is possible that a particularly severe accident would breach the containment structure leading to unattenuated release of gaseous and particulate radionuclides suspended in the containment volume. Determination of the probability and consequences of such extreme events is an important aspect of the safety analysis of commercial nuclear reactors [2-4].

Confinement systems, on the other hand, control the flow of effluents produced in a reactor accident so that they pass out of the confinement along prescribed pathways equipped with features to attenuate the release of radioactivity. The escape of gaseous radionuclides—xenon, krypton, and especially, in the case of the production reactors, tritium—is accepted in return for assurance that even in a very severe accident there will be significant mitigation of the release of all other radionuclides. The damage to health and property caused by release of these other radionuclides is thought to be far greater than would be caused by release of the gaseous radionuclides [5].

The confinement concept is acceptable in principle provided the confinement has been designed to accommodate sufficiently severe reactor accidents (see the discussion of commercial reactor siting criteria in Appendix H). One advantage of the production reactors is that their site boundaries are distant from the reactors, so significant dilution of escaping, gaseous radionuclides is possible during an accident. There is no compelling evidence that mere adoption of the containment concept would substantially improve the safety of the production reactors. The confinement concept is equally viable. Indeed, there is much current interest, particularly in Western Europe, in modifying the containments of some commercial nuclear power plants so that during severe reactor accidents they would function much like confinements [6-8]. The "controlled venting of containments" was also proposed in the reactor safety study of the American Physical Society in 1975 [39].

It is in the implementation of the confinement concept at the production reactors rather than the concept itself that is cause for concern. These concerns involve the following:

- Performance of mitigation systems in the production reactor confinements during severe accidents,
- Capabilities of the confinement systems at the production reactors to sustain loads arising during severe accidents, and
- Confinement of routine and accidental discharges of contaminated liquid effluents.

The last of these concerns is discussed in Chapter 2 and in Appendix E. Concerns over the abilities of the production reactor confinements to mitigate radionuclide releases adequately and concerns over the ability to sustain severe accident loads are discussed here.

At both the Savannah River reactors and N Reactor, attenuation of radionuclide release is accomplished by a filtration system. The confinement systems at Savannah River are continuously ventilated through a filtration system during normal and accident conditions [9]. The N Reactor system is more complicated. In the event of a rupture of the pressurized coolant system at N Reactor, the confinement system is depressurized through large relief vents. These vents are not equipped to attenuate the release of any radionuclides suspended in the effluent. Automatic closure of the vents and redirection of the flow through cells equipped with filters are controlled by timers. The timing sequence is based on hypotheses concerning large breaks in the reactor coolant system [10].

In the committee's view, it has not been clearly established that initial effluents produced during the course of an accident at N Reactor will be sufficiently free of radionuclides so that direct venting of confinement pressure will make no significant contribution to the radionuclide release from the plant. The various manipulations of flow in the N-Reactor confinement were developed by hypothesizing a particular accident history, yet the committee is unconvinced by the available analyses that all accidents will conform to this history. For example, accidents initiated by the rupture of a single pressure tube and involving core degradation under pressure may not conform to the design hypotheses. Difficulties have been encountered with the timing sequence for manipulating confinement flow for N-Reactor accident scenarios only modestly different than those used to design the system [11].

Even if the confinement flow manipulations at N Reactor are satisfactory, questions remain about the ability of the filtration systems both at N Reactor and at Savannah River to mitigate radionuclide releases throughout an accident. The filtration systems of both confinements have been designed with a radionuclide release in mind that conforms to the so-called "TID Source Term," [12] which involves the following assumptions:

Radionuclide	*Release During an Accident*
Noble gases	100% (includes ^3H)
Iodine	50% (as I_2)
Other radionuclides	1% (as particles)

Based on this source term, the filtration systems at Savannah River and N Reactor have been constructed to consist of three elements:

1. A demister to remove suspended water droplets from the effluent,
2. A high-efficiency-particle-air (HEPA) filter to remove radionuclides in particulate form from the effluent, and
3. A charcoal bed to absorb gaseous iodine from the effluent.

The designs have thus used the TID Source Term as a prescription of not only the magnitude of the overall radionuclide release during a severe reactor accident, but also as a prescription of the chemical and physical forms of the released radionuclides.

Unfortunately, the TID Source Term was not formulated on a technical foundation that makes it suitable for the design of filtration systems. The TID Source Term was created at a time when very little was known about radionuclide releases during reactor accidents. The source term was propounded as a supposedly conservative upper bound based not on factors important for the design of filtration systems but rather in relation to the potential consequences of radionuclide releases. At the time that this source term was propounded, the consequences of releases of iodine to the environment were thought to be most severe, so extensive release of iodine was hypothesized in order to compensate for ignorance concerning the release of other radionuclides. The gaseous form of iodine was prescribed based on a belief that this chemical form would be the most difficult to attenuate.

Research on severe accident source terms has been extensively pursued in connection with the regulation of commercial nuclear power reactors [2,3]. This research has shown that source terms produced during reactor accidents will be quite different than hypothesized in the TID Source Term. The following points are among the more important research findings:

1. Iodine release from degrading reactor fuel will be nearly complete and may not be as iodine gas.
2. Release of radionuclides in particulate form can be greater than the 1 percent release hypothesized in the TID Source Term.
3. Nonradioactive materials in particulate form will make great contributions to the effluent produced in a reactor accident.

Recently, tests of radionuclide release from heated fuel from

TABLE B.1 Radionuclide Masses in Production Reactor Fuels

Element	Mass (kg) in SRP Mark 16-B Fuel [18]	SRP Mark 31-A Fuel [18]	N Reactor Fuel [33]
Bromine	6.39	0.54	0.21
Rubidium	6.16	0.50	0.37
Tellurium	5.20	0.87	3.69
Iodine	2.43	0.56	2.20
Cesium	36.90	3.67	22.08

the Savannah River reactors have been conducted [13]. These tests show the following:

1. Iodine release is nearly complete.
2. Cesium, which in the confinement atmosphere will form particulates, is extensively released from overheated fuel.
3. Iodine was found to deposit in association with cesium in the test apparatus as though the iodine were in the form of CsI rather than I_2 gas.

These findings are consistent with the findings of past tests and models of radionuclide release from metallic fuels [14-16]. Similar tests of radionuclide release from N-Reactor fuel have not been conducted, but are planned [17]. During severe accidents at N Reactor, the fuel will get at least as hot and will be exposed to thermal hydraulic conditions at least as severe as Savannah River reactor fuel [10], so similar results can be expected. These findings show iodine could be in the particulate form, CsI, rather than in a gaseous form (I_2 or HI) and could collect on the filtration unit rather than the charcoal bed. But, of greater concern, release of significant amounts of cesium, present in some fuels in masses on the order of tens of kilograms (see Table B.1), substantially increases potential particle loadings on the filters, threatening their integrity.

There are many other sources of nonradioactive aerosol particles that could develop during an accident at these plants:

- Particulates produced from the fuel and clad during heatup in steam,

- Smoke produced by combustion of organic materials within the reactor confinements [19],
- Coke formation when mixtures of H_2, CO, CO_2, and H_2O formed within the hot moderator block of N Reactor cool [20], and
- Boric acid produced by steam corrosion of boron carbide [21].

Accidents at Savannah River that progress to the point at which core debris interacts with concrete [22] would yield formidable quantities of aerosols [23]. Accidents involving the U_3O_8-Al powder metallurgical fuel currently under development (see the discussion of cermet fuel in Chapter 2 and in Appendix F) would generate significant amounts of nonradioactive aerosol if metallothermic reduction of the U_3O_8 by Al occurs [24].

Analyses of the loading of particle filters during hypothetical accidents at the Savannah River reactors and at N Reactor have not included considerations of nonradioactive particulates, whereas the contributions of nonradioactive sources to aerosols suspended in the reactor containment atmosphere are usually found in the analyses of accidents at commercial nuclear power plants to be overwhelmingly larger than contributions made by released radionuclides [35]. These nonradioactive aerosols, together with greater than expected releases of radionuclides in particulate form, would threaten to clog and rupture the filters. If the filter system were to rupture, a large release of radionuclides from the production reactors could occur.

Larger releases of radionuclides in particulate form might also place unexpected heat loads on the filter systems. The design basis for heat loads on filters at N Reactor assumes that only 0.1 percent of the core inventory of fission products that would be released in nonvolatile, particulate form will come to reside on the filters [10].

HEPA filters used in the production reactor confinements are quite efficient at removing particulate material from gas streams, but they have limited capacities for handling particulate loading. The pressure drop across an HEPA filter increases approximately linearly with loading up to a critical value. As loading of the filter with particulates increases beyond this critical value, the pressure drop increases approximately exponentially with loading as shown in Figure B.1. Studies of a variety of HEPA filters suggest that filter loadings of 1.5 to 4.0 kg/m^2 mark the onset of rapidly increasing pressure drop [25], which leads to failure of the filters.

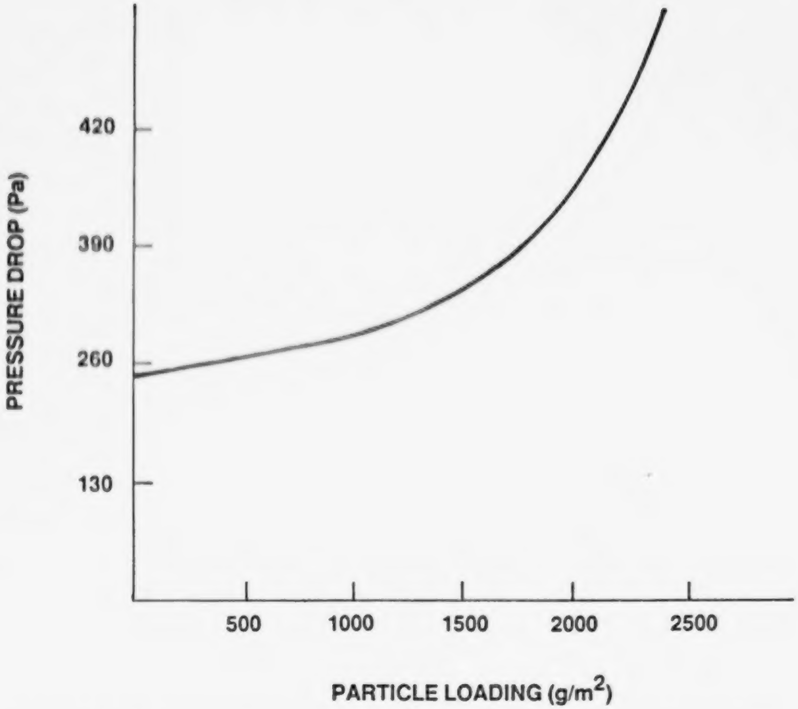

FIGURE B.1 **Pressure drop across a standard American design HEPA filter as a function of particle loading [25].**

This implies that the filters in N Reactor could sustain loads of 84 to 120 kg, at most. More conservative analyses by the contractor at N Reactor suggest that loadings of no more than 10.8 kg would be defensible [30].

Filter systems in the Savannah River reactors could sustain loads of 32-48 kg of particulates in each filter compartment. In fact, analyses of pressure drop data obtained following the source rod accident at K Reactor in 1970 suggest that loads no larger than this could be tolerated [26]. Data from this accident do, however, show that particulate may also be collected on the moisture separators, and the capacities and efficiencies of these devices as particle traps are unknown.

At N Reactor, the mitigation of releases provided by the filters would be augmented by confinement sprays. It is claimed that sprays would remove no less than 98 percent of the particulates from the N-Reactor confinement. Upgrades for the Savannah River reactors would make sprays available for operator actuation [18]. Sprays in commercial reactor containments are credited with high mitigation capabilities [2,3]. Confidence in the mitigative capabilities of sprays at the production reactors has been criticized by previous reviewers [27]. Indeed, the analyses of spray capabilities at N Reactor must be viewed with some skepticism. The analyses were performed using a commercial reactor computer code (CORRAL) [28] that has been abandoned in the safety analysis of commercial reactors in favor of more sophisticated models [2,29]. Parameterization of the CORRAL code was based on tests appropriate for containment [31] rather than the once-through flow of the N-Reactor confinement. The analyses also hypothesized particulate sizes of 5 to 15 μm. But the efficiency of spray decontamination of particulate-laden gases is a strong function of aerosol particle size, as shown in Figure B.2. Analyses for commercial reactors, including core degradation experiments [2,3], and the experience of the Chernobyl accident show that much finer particles, which are less easily removed by sprays, could form during reactor accidents. Experimental and analytical work, using analyses that reflect the unique flow conditions that exist in the N-Reactor confinement, needs to be done to determine whether the low collection efficiencies of the confinement sprays for small particle size aerosols would still yield large factors of decontamination. These analyses should also take into account that spray coverage of the N-Reactor confinement is not uniform. If particle-laden gases bypass the sprayed regions, they will pass unattenuated to the filter systems.

The filter systems used at both the Savannah River reactors and N Reactor are not especially strong. Pressure pulses sufficient to rupture the filters are estimated to be 10 to 12 inches of water (0.5 psi) [9,30,37]. These estimates are consistent with predictions found in the literature [34]. Such pressure pulses would be quite likely to develop in the course of severe reactor accidents. As discussed in Chapter 2, hydrogen combustion, which need not be intense enough to produce shock waves, could still produce pressure pulses sufficient to rupture the filters and even the charcoal beds used for trapping iodine. Parametric calculations for N Reactor based on the modest hydrogen generation rates of the hypothetical accident indicate the possibility of pressure pulses that threaten filter integrity [38]. Thus, bulk rupture might render the

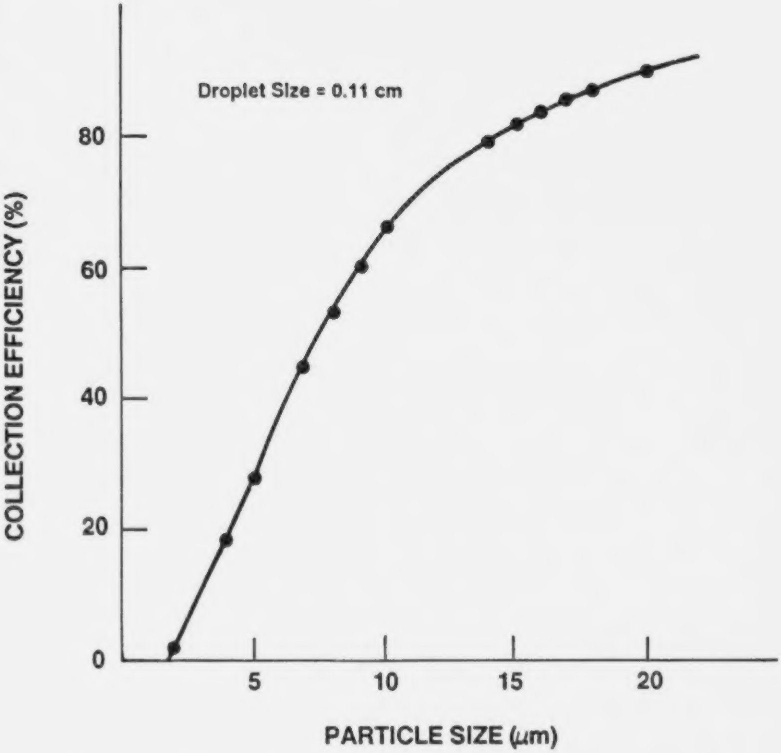

FIGURE B.2 Particle collection efficiencies for fog spray droplets at N Reactor as functions of particle size. (Calculations were made using models from reference 29. Droplet size data are from reference 10.)

confinement filtration system, if not the confinement itself, incapable of attenuating radionuclide releases. Bulk rupture caused by pressurization from a hydrogen deflagration or detonation might be especially troublesome since such an event could also cause deposited, radionuclide-bearing aerosols to be resuspended [35,36].

NOTES

1. The first use of the term "defense in depth" as applied to the safety of nuclear power plants seems to have been made in the period 1964-1965, when the Regulatory Staff of the Atomic Energy Commission was formulating the requirements

for Technical Specifications for licensed plants. A logical system was defined according to which each operating variable (e.g., reactor power) had assigned to it an operating range, a safety limit that should not be exceeded because this would lead to some kind of damage, and a trip point that would lead to correction or shutdown before the safety limit is exceeded in transients. This was called a defense in depth.

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Appendix C
Documents on Power Operating Limits at Savannah River Plant

NATIONAL RESEARCH COUNCIL
2101 Constitution Avenue Washington, D.C. 20418

COMMISSION ON PHYSICAL SCIENCES,
MATHEMATICS, AND RESOURCES

COMMISSION ON ENGINEERING AND
TECHNICAL SYSTEMS

COMMITTEE TO ASSESS SAFETY AND
TECHNICAL ISSUES AT DOE REACTORS

March 9, 1987

The Honorable John S. Herrington
The Secretary of Energy
U.S. Department of Energy
Washington, D.C. 20585

Dear Secretary Herrington:

On May 13, 1986, you requested that the National Academy of Sciences and the National Academy of Engineering form a committee to perform an assessment of the safety of the Department's major production and research reactors. The Committee's effort is underway, with a principal focus initially on the production reactors at Savannah River and the N-reactor at Hanford. I am writing in response to a request by Under Secretary Salgado at our initial meeting that we bring any matters of immediate concern promptly to the Department's attention.

As you know, the allowable reactor power limits for the Savannah River reactors were decreased last November based on a reevaluation of the adequacy of the emergency cooling systems in a severe loss of coolant accident. As a result, the Committee has examined whether acceptable fuel temperature is likely to be maintained in such an accident. I am writing to set out the Committee's concerns on this matter.

Based on interviews over several days with Du Pont and DOE staff and the review of documents made available to the Committee, we are not able to conclude with confidence that significant core damage would be avoided if there were a severe loss of coolant accident while the reactors are operating at the currently established reduced power limits. In our view, the reactors should only be operated at power levels at which it can be convincingly demonstrated that there will be adequate cooling of the fuel over the entire duration of the transient.

We recommend that the Department examine this matter immediately and establish an aggressive course of action for the short term to insure the safety of the reactors in a severe loss of coolant accident. In this connection, both DOE and DuPont should obtain expert outside guidance and advice on the matter in a vigorous and timely fashion. Over the longer term increased

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resources should be devoted to developing a more thorough understanding of the behavior of the reactors in a severe loss of coolant accident. Finally, prototypic experiments and more rigorous analyses should be undertaken to explore the phenomena that affect reactor behavior in such a situation.

The Committee will review this issue further and will discuss it in more detail in its subsequent reports.

Sincerely,

Richard A. Meserve

Richard A. Meserve
Chairman

Department of Energy
Washington, DC 20585

March 12, 1987

Dear ~~Mr. Meserve~~ Dick:

The Department has received your letter of March 9, 1987. I appreciate your early expression of concern about the adequacy of emergency core cooling systems, should there be a severe loss of cooling accident at one of the Savannah River reactors.

I have asked Assistant Secretary Mary Walker to establish an action plan to examine this matter as expeditiously as possible. To aid us in understanding your particular concerns, it would be helpful if members of your committee could brief DOE staff as soon as possible. Ms. Walker will contact you to arrange the briefing.

Yours truly,

Joseph F. Salgado
Under Secretary

Richard A. Meserve, Chairman
Committee to Assess Safety and
Technical Issues at DOE Reactors
National Research Council
2101 Constitution Avenue, N.W.
Washington, D.C. 20418

Department of Energy
Washington, DC 20585

May 5, 1987

Mr. Steve N. Blush
Study Director
National Academy of Sciences
Office 271
2101 Constitution Avenue
Washington, DC 20418

Dear Mr. Blush:

This letter responds to your telephone request to Marilyn Smith for information on the DOE's review of the power levels selected for the Savannah River Reactors to meet the National Academy of Sciences' and the National Academy of Engineering's concerns expressed in Mr. Meserve's March 9, 1987, letter to Secretary Herrington. My office organized an effort under the leadership of W. W. Kinney, Acting Branch Chief of the Reactor Safety Branch in the Office of Nuclear Safety, who was assisted by a team of three international experts in emergency core cooling. The team's report is documented in a letter dated April 6, 1987, from P. North, Manager, Nuclear Reactor Research and Technology, EG&G Idaho, Inc., to R. E. Tiller, Director, Office of Special Programs, DOE, Idaho Operations Office (see enclosure 1).

Based upon the results of the team's assessment of the methodology used to establish the current permissible power level, we consider that the current operating power levels include conservatisms with respect to possible fuel damage in the event of a large break loss-of-coolant accident. The EH staff also had a number of recommendations (see enclosure 2) with respect to the continuing analysis of the power limits for the reactors. These recommendations are being implemented by the Office of the Assistant Secretary for Defense Programs. We will be pleased to provide you or the committee with additional information as necessary.

Sincerely,

Mary L. Walker
Assistant Secretary
Environment, Safety and Health

2 Enclosures

INEL
Idaho National Engineering Laboratory

April 6, 1987

Mr. R. E. Tiller, Director
Office of Special Programs
Idaho Operations Office - DOE
Idaho Falls, ID 83401

REVIEW OF METHODOLOGY USED IN SAFETY ANALYSIS OF THE SRL PRODUCTION REACTORS

Dear Mr. Tiller:

On March 24 and 25, 1987, Mr. John Liebenthal and Dr. Victor Ransom of EG&G Idaho, Inc. and Mr. Kenneth Moore, of Scientech, accompanied Mr. William Kinney, of DOE-EH, to Savannah River Laboratories for the purpose of reviewing safety analysis that had been used to set conservative operating power levels for the SRL production reactors.

A summary of the review with conclusions and recommendations is attached. The team agreed that the current operating power levels included significant conservatisms with respect to possible fuel damage in the event of a large break LOCA. However, significant uncertainties exist with respect to the cooling process under low ECCS flows. In view of these uncertainties, it is recommended that the limiting case cooling analyses be completed and reviewed. Additional recommendations are made relative to future efforts to quantify safety margins.

Very truly yours,

JM Broughton/for

P. North, Manager
Nuclear Reactor Research
and Technology

amj

Attachment:
As Stated

cc: W. Kinney, DOE-EH
K. Moore, Scientech

EG&G Idaho, Inc. P.O. Box 1625 Idaho Falls, ID 83415

Rec'd by OSI APR 87

EG&G Idaho

INTEROFFICE CORRESPONDENCE

Date: March 30, 1987

To: J. O. Zane

From: J. L. Liebenthal, K. V. Moore, and V. H. Ransom

Subject: REVIEW OF THE METHODOLOGY USED IN SAFETY ANALYSIS OF THE SRL PRODUCTION REACTORS - Rans-18-87

Introduction

A technical review team consisting of Mr. John L. Liebenthal and Dr. Victor H. Ransom, EG&G Idaho, and Mr. Kenneth V. Moore, Scientech, accompanied Mr. William Kinney, DOE-EH, to The Savannah River Laboratories on March 23, 1987 and met with SRL personnel on March 24 and 25, 1987. The purpose of our visit was to review the rationale used to analyze the response of the SRL production reactors to a large break LOCA. At the time of the review, the reactors had been reduced to approximately 50% power levels as a result of safety concerns expressed by the National Academy of Science. The objectives of the meeting were to review the methodology used to establish the current operating power levels, to review the procedure proposed for establishing power levels for operation over a 90 day period, and to comment on the approach proposed for establishing higher operating power levels to be implemented at the end of 90 days. The findings and opinions of the EG&G Idaho and Scientech personnel are summarized in this report.

The National Academy of Science review group had previously expressed concern with regard to the ability to cool the reactor fuel and/or target assemblies using emergency core cooling in the event of a 200% double offset shear cold leg break accompanied by an additional failure of one ECC circuit, i.e. only one of the three ECC cooling circuits is assumed available. The safety criterion that had been used to establish operating power levels, prior to review by the National Academy of Science, was that the maximum fuel temperature could not exceed the melting point of the aluminum cladding. The current operating power levels were set using a more conservative criterion that no net boiling would occur in any assembly, i.e. that the coolant outlet temperature would be 100 degrees C with no net steam production. This criterion was selected in order to avoid the possibility of counter current flow limiting (CCFL) at the top of the fuel and target assemblies, which was postulated by the National Academy of Science as a mechanism that could lead to fuel heat up and damage.

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System Description

A brief description of the Savannah River Reactors will be given here in order to facilitate explanation of the conclusions that are subsequently described. The reactors have aluminum clad enriched U235 fuel in the form of concentric cylinders. Each fuel assembly consists of three concentric cylinders that are approximately 4 inches in diameter and 12 feet in length. Target assemblies consist of two concentric aluminum clad cylinders of either U238 or Al-Li alloy that are also approximately 4 inches in diameter and 12 feet in overall length, but are made up of a train of 8 inch long slugs. The core consists of 600 fuel and target assemblies arranged in a hexagonal array consisting of a fuel assembly surrounded by six target assemblies. The fuel and target assemblies are both convectively cooled by circulation of the heavy water which also is the moderator. The heavy water is circulated through the core by six symmetrically arranged coolant loops each containing a pump and heat exchanger. Three of the cooling loops include provision for injection of light water as emergency core cooling in the event of a LOCA. The topology of the cooling system is such that the cooling water is introduced at the top on the reactor through a plenum and flows down through the 600 fuel and target assemblies and exits into the reactor tank. Pump suction lines are attached at the tank bottom and the coolant is pumped through heat exchangers, cooled by river water, and reintroduced at six equally spaced circumferential locations into a distribution plenum that is 8.5 inches high and 10 feet in diameter. The reactor power level is controlled by control rods that enter the reactor core from the top.

Postulated Accident Scenario

The postulated accident of concern is a large break LOCA in which one of the six reactor inlet cooling pipes suffers a double offset shear break. In particular, one of the three inlet pipes that contain ECC injection provision is assumed to break. Moderator/coolant is lost out the break due to draining of the upper plenum and due to pumping out the pump discharge line (there is little or no blowdown since the reactor operates at low pressure). With one ECC system supplying coolant to the reactor, the system reaches an equilibrium condition after approximately one minute in which the tank level has dropped to a depth of approximately 18 inches and air from the tank is entrained into the pump suction lines (air flows into the reactor vessel through vacuum breaker valves). The mixture of water and entrained air (estimated by DuPont to be 5% air by volume) is pumped to the upper plenum where flow stratification occurs and water flows into the fuel and target assemblies as a result of gravitational head only. Water flows from the plenum out the broken inlet pipe and air will flow in or out the break depending on the amount of air entrained in the assembly coolant flows. Additional water and air are discharged from the system out the broken pump discharge line. The point of minimum water level in the upper plenum, and consequently also the point of minimum coolant flow, occurs at a point between the break and the center of the core. Thus, the limiting assembly, i.e. the one with the greatest ratio of decay heat power to coolant flow, also occurs nearby. The limiting assembly establishes the maximum reactor operating power level since the decay heat power is almost in direct proportion to the operating power level.

There are two time periods after occurrence of a LOCA in which ability to cool the fuel and/or target assemblies is a concern. The first occurs in the 0 to 12 s time period and is the result of a coupled fluid-thermal instability in which fuel heat up occurs due to reduced coolant flow under conditions of net vapor production. The second period occurs between 40 - 60 s after LOCA occurrence and is the point at which the decay heat power to cooling capacity ratio is a maximum and thus is the time at which the maximum fuel/target temperature and coolant outlet temperature result in the limiting assembly. When the no bulk boiling criterion is employed, the most limiting condition occurs during this second period. Thus, most of the discussion in this review concerns the methods used to predict the coolant flow rate to the limiting assembly and the ability of the resultant coolant flow to effectively cool the assembly.

Methodology Assessment

The methodology that was used to establish the current permissible power levels is based on empirical and analytical estimates for the decay heat power, available coolant flow, and heat transfer conditions in the limiting assembly. The methods used to establish each of these factors will be discussed relative to the degree of conservatism that is included. The term "best estimate" is used in the sense that the most probable result would be predicted. The term "conservative" is used herein to mean more conservative than "best estimate."

The decay heat power was established using the 1973 ANS standard which is a best estimate value. Present NRC practice is to obtain a conservative estimate for decay heat power is to use a 1.2 multiplier on the ANS standard. Less restrictive best estimate methods for decay heat power are currently under review by NRC, but are not yet approved. An additional difference between the SRL reactors and the basis for the ANS decay heat standard is that the SRL reactors breed Pu239 and contain fission products of Pu239 while the ANS standard is based on fission of U235. DuPont stated that they had looked at this effect and concluded that the ANS decay heat model was conservative. The method used to estimate decay heat power is a best estimate model.

The method used to establish the coolant flow to the limiting assembly was based on empirical data gathered in full scale tests of the reactor system with one coolant pump not operating. These data were used to establish flow coefficients for the SRL FLOOD84 code which in turn was used to predict flows to individual assemblies under conditions other than those corresponding to the test conditions. This procedure does not include additional conservatisms to account for variations due to data inaccuracy, effect of break location on coolant loss, effects of uncertainties in the amount of entrained air in the recirculated flow, etc. Therefore, this model results in a best estimate prediction for the coolant flow to the limiting assembly.

The degree of conservatism of the method used to estimate the channel-to-channel flow division and the associated cooling effectiveness is difficult to judge. The experimental data taken on channel-to-channel flow division had a very wide scatter and a two-sigma low value was used even though lower values of flow to the outermost channel were actually measured. The cause of the wide dispersion in the measured flows is not known and could be due to experimental procedure or a real effect. Thus, the use a two-sigma low value may not result in a conservative estimate for the low flow since approximately 60 assemblies are located in the area of low upper plenum level. This exceeds the number of data points used to establish the two-sigma low limit and it is possible that the lowest measured flow could occur. A more conservative estimate for the percent of assembly flow going to the outer annuli would be the lowest value measured, i.e. approximately 10% rather than 12%. The method that was used yields an estimate for the flow to the outer channel of the limiting assembly that is between a best estimate and a conservative value.

The degree of conservatism in the heat transfer calculation used to determine the bulk temperature rise is also difficult to estimate. The flow in the outer annuli is undoubtedly significantly maldistributed in view of the wide scatter in the channel-to-channel flow division. Maldistribution will result in nonuniform heat transfer in contrast to the uniform assumption used in the cooling calculations. For a fixed coolant flow to the limiting channel, the most probable situation is that maldistribution will be present which will result in locally higher wall temperatures than would be the case with uniform heat transfer. Most likely, any energy not absorbed by the coolant in the limiting channel will be distributed to the other cooling channels by conduction without excessive fuel temperature. However, heat transfer analyses for the most limiting configuration, the MK31A target assembly, under conditions of coolant maldistribution were not presented. The assumption of uniform heat transfer in the limiting channel yields fuel temperature estimates that are less conservative than a best estimate model that includes maldistribution.

Additional conservatism was added explicitly by reducing the maximum operating power level determined using the no boiling in the limiting assembly criterion, by an extra 5% and through the use of simple power ratios established by a detailed limits analysis using the previous no fuel damage criterion. DuPont estimated that an additional 18 to 34 percent in power margin existed as a result of this procedure.

Conclusions

In summary, review of the conditions under which the SRL production reactors are currently being operated indicated that power margins probably exist compared to a best estimate calculation of the limiting power. However, uncertainties with respect to the heat transfer process in the limiting assembly continue to exist. Specifically, uncertainties with respect to the heat transfer effectiveness in a channel that is greater than 90% void and under conditions of circumferential coolant flow maldistribution. Neither experimental data nor detailed calculations were presented that demonstrated that cooling adequate to prevent local fuel damage is achieved. Related heat transfer studies on the ability of the fins on the MK16 fuel to conduct the decay heat between adjacent cylindrical fuel elements were presented in order to support the contention that adequate cooling could be achieved. Mental extrapolation of these results to the case of a MK31A target assembly, in which assumed maldistribution of the coolant would require the decay heat to be conducted circumferentially over approximately one half of the circumference of the target ring, indicated that clad melting probably would not occur.

In our opinion, the analysis that has been made to support the current operating power levels contains ad hoc procedures, lacks consistency, and lacks detailed independent review. While we believe that the current operating power levels contain adequate conservatisms, based on the material presented by the DuPont personnel, we would not recommend any increase over the current operating levels until a consistent analysis of the decay heat removal process is completed, documented, and reviewed.

Recommendations

In addition to the comments that have been made, the following specific recommendations are offered:

1. The mechanisms that are responsible for the large variation in the measured channel-to-channel flow distribution need to be understood. In particular, some of the variation may be due to manner in which the experiments were conducted which introduces unnecessary penalty. Conversely, if the observed variations are

due in some manner to the physical assembly, then the minimum observed flow may be real and would, therefore, be the appropriate value to use rather than a two-sigma low.

2. The accompanying circumferential flow distributions in the coolant channel need to be established or a worst case analysis needs to be incorporated into the limits analysis methodology.

3. A detailed conduction and convection heat transfer analysis of the fuel and target elements under conditions of assumed coolant maldistribution should be carried out as soon as possible.

4. Sensitivity studies need to be incorporated into the methodology used to set power levels so that safety margins can be better quantified.

5. An independent safety study should be undertaken by an independent team skilled in the methods used in LWR safety.

6. The detailed analysis of plenum flows should be continued.

7. Development of an integrated safety analysis capability based on existing LWR technology is strongly encouraged. The TRAC-BD1 computer code is recommended as a starting point since this code already has many of the modeling features needed to simulate the SRL production reactors.

John L. Liebenthal
Manager of Chemical and Process
Engineering

Kenneth V. Moore
Scientech, Inc.

Victor H. Ransom
Scientific and Engineering Fellow
Nuclear Reactor Research and Technology

cc: R. L. Benedetti
P. North

MAY 4 1987

EH-1

EH Review of Methodology Used by Savannah River Operations Office (SR) in Meeting No Bulk Boiling Criterion in the SR Production Reactors

S. R. Foley, DP-1

Reference: Memorandum from Mary L. Walker to Joseph F. Salgado, "Resolution of Concerns about Savannah River Emergency Core Cooling Systems," dated April 1, 1987

The purpose of this memorandum is to present the conclusions and recommendations of the EH team which performed an independent review of the safety basis for the reduced power levels for the Savannah River Plant (SRP) production reactors established by SR to satisfy the "no bulk boil upon loss-of-coolant-accident (LOCA)" criterion.

On March 20 and 21, 1987, du Pont prepared, and SR approved, power levels intended to assure that no bulk boiling of emergency cooling water would occur under double ended pipe break LOCA conditions with only one of three emergency core cooling systems (ECCS) lines functional. On March 24 and 25, 1987, the subject review was performed by technical experts Dr. V. H. Ranson and Mr. J. L. Liebenthal of EG&G Idaho, and Mr. K. V. Moore of Scientech, Idaho Falls, and Team Leader W. W. Kinney of the Office of Nuclear Safety.

Based upon results of the team's assessment of the methodology used to establish the current permissible power level, EH considers the current operating power levels to include some conservatisms with respect to possible fuel damage in the event of the large break LOCA. EH recommends that any consideration of an increase in power over the current operating levels be based upon the completion, documentation, and review of a consistent analysis of the decay heat removal process under the low ECCS flow which would be created by the design basis LOCA. Specific EH staff recommendations and the letter and report prepared by the technical experts are attached.

It is requested that EH be provided detailed information on the basis for any recommended consideration of an increase in power levels for our review and concurrence.

DISTRIBUTION:
Subject
EH-30 RF
ONS RF
NFSD RF
Kinney RF

Original signed by:

Mary L. Walker
Assistant Secretary
Environment, Safety and Health

5/4

2 Attachments

EH-332.1:WWKinney:srh:353-5087:4/17/87:rewritten 4/22/87:
rewritten 4/29/87:rewritten 5/4/87

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Attachment 1

EH Staff Recommendations

The following recommendation applies only to the establishment of the power level limits based on the criterion of no bulk boiling.

1. Have an independent group conduct a review of the technical bases for future increases in the power limits which would comply with the no bulk boiling of emergency cooling water criterion. (See team recommendation 5.)

The following four recommendations, with appropriate scopes, apply both to: (1) the establishment of power limits based upon the criterion of no bulk boiling, and (2) the subsequent reestablishment of power limits based upon the criterion of no assembly damage with limited boiling.

2. Determine the mechanisms that are responsible for the large variation in the experimentally measured channel-to-channel flow distribution under low coolant flow conditions. (See team recommendation 1. for further information.)

3. Either establish the circumferential flow distributions in the coolant channel under the low flow conditions or incorporate a worst-case analysis into the limits analysis methodology. (See team recommendation 2. for further information.)

4. Carry out detailed conduction and convection heat transfer analysis of fuel and target assemblies under low coolant flow conditions with assumed coolant maldistribution. (See team recommendation 3.)

5. Incorporate sensitivity studies into the methodology used to set power levels to permit better quantification of safety margins. (See team recommendation 4.)

The following two recommendations apply only to the reestablishment of power limits based upon the criterion of no assembly damage with limited boiling.

6. Have an independent group skilled in methods used in commercial light water reactor (LWR) safety analyses conduct an independent study of the cooling process under the design basis LOCA conditions to verify the du Pont analyses. (See team recommendation 5.)

7. Develop an integrated safety analysis capability based on existing LWR technology. Use the TRAC-BDI computer code as a starting point since this code already has many of the modeling features needed to simulate the SR Plant production reactors. (See team recommendation 7.)

EH also supports the continuation of the detailed analysis of plenum flows currently being conducted by du Pont. (See team recommendation 6.)

DOE F 1325.8
(12-84)
United States Government **Department of Energy**

memorandum

DATE: APR 2 4 1987

REPLY TO ATTN OF: EH-332.1

SUBJECT: ONS Review of the Savannah River (SR) Action Plan to Respond to Concerns About Emergency Core Cooling System Capability of SR Production Reactors

TO: Charles D. Simpson, DP-13

On April 16, 1987, you provided to the Office of Nuclear Safety (ONS) for our review a copy of a draft of DuPont's "Action Plan to Respond to Concerns About SR Emergency Core Cooling Capability" dated April 14, 1987, which was sent to John L. Meinhardt, DP-10, by R. L. Morgan, Savannah River Operations Office, for concurrence. Under the leadership of W. W. Kinney of my office, we put together a team of ECC experts including Dr. V. H. Ransom and Mr. J. L. Liebenthal of EG&G, Idaho, and Dr. L. J. Ybarrando and Mr. K. V. Moore of Scientech to review the DuPont action plan.

These gentlemen were familiar with the concerns about the capability of the emergency core cooling systems (ECCS) of the SR production reactors. Dr. Ybarrando participated in the EH Design Review of the reactors in November 1986, and reviewed the design and analysis of the ECCS. Dr. Ransom and Messrs. Liebenthal and Moore on March 24-25, 1987, assisted ONS in reviewing the methodology used by SR in meeting the "no bulk boiling of emergency cooling water upon LOCA" criterion imposed by Under Secretary Salgado on March 20, 1987.

In order to meet your requirements for comments by COB on April 24, 1987, we have attached an advance copy of the report resulting from the review of the DuPont action plan. The comments are substantive, and we request that the plan be appropriately revised in consideration of the comments. We conclude that the action plan does not provide a clear definition of the objectives, strategy, and details needed to provide a well planned and executed solution to the current ECC questions. We underscore the recommendation that further development of the action plan be assisted by individuals experienced in resolving similar problems.

As previously mentioned, on March 24-25, 1987, Dr. Ransom and Messrs. Liebenthal and Moore, along with W. W. Kinney of ONS, reviewed the methodology used by SR in meeting the "no bulk boiling of emergency cooling water upon LOCA" criterion. The EH staff recommendations which resulted from this review and the report providing the team's conclusions and recommendations also pertain to the action plan and should be considered in any revision of the subject action plan. These are being transmitted separately relative to the selection of the current power levels. However, in order to facilitate actions on the subject action plan, we are attaching copies of these documents.

With regard to the SR "ECS Action Plan/DOE Overview Activities," which was included with the DuPont action plan, ONS wants to reinforce SR's intent to have independent groups conduct reviews of the technical bases for any considerations of future increases in power limits under the action plan.

Thank you for the opportunity to review the subject action plan.

Robert W. Barber
Director
Office of Nuclear Safety

Attachments

21 APRIL 1987

Mr. R. E. Tiller, Director
Office of Special Programs
USDOE
Idaho Operations Office
Idaho Falls, Idaho 83401

Subject: Review of Action Plan for Savannah River ECCS, YBR-253-87

Dear Mr. Tiller:

As requested, Dr. V. Ransom and Mr. J. Liebenthal of EG&G, Idaho and Mr. K. V. Moore and I of SCIENTECH have reviewed the Savannah River Action Plan on severe LOCA transient uncertainties. Particular attention was placed on the actions planned by Du Pont for the ECCS related technical issues.

Our team evaluation of the action plan is contained in Attachment 1 to this letter.

If we can be of any further service in this important matter, please contact us.

Sincerely,

L. J. Ybarrondo, PhD
President

cc: Mr. R. D. Lease, DOE-ID
Mr. R. W. Barber, DOE-HQ
Mr. J. O. Zane, EG&G, Idaho
Dr. V. Ransom, EG&G, Idaho
Mr. J. Liebenthal, EG&G, Idaho
Mr. K. V. Moore, SCIENTECH, Inc.

ATTACH. 1 TO YBR-253-87: SRP ACTION PLAN 4/23/87

1.0 Introduction

The team was asked to evaluate a 14 April 1987 draft entitled, "ACTION PLAN TO RESPOND TO CONCERNS ABOUT SR EMERGENCY CORE COOLING CAPABILITY ". The subject action plan presents a program to respond to concerns about the effectiveness of emergency cooling following a severe LOCA in a Savannah River Plant (SRP) production reactor. These concerns were expressed by the Design Performance Review team in November 1986 and the National Academy of Sciences SRP review panel in a March 9 letter to Secretary Herrington.

The Design Performance Review team noted that the accident analysis tools in use are outdated and recommended they be completely modernized immediately. Further, if the ongoing PRA showed that the ECCS was a critical component of safety risk then the ECCS should be tested in the C reactor or another suitable place. The National Academy of Sciences review panel stated, " We are not able to conclude with confidence that significant core damage would be avoided if there were a severe loss-of-coolant accident while the reactors are operating at the currently established reduced power limits." The conclusions of both panels are similar. The subject action plan is intended to address the above concerns.

The power of the SRP reactors was restricted further on March 20, 1987. On March 24 and 25, 1987, Dr. Victor Ransom, Mr. Kenneth Moore, and Mr. John Liebenthal, along with Mr. William Kinney of Environment, Safety, and Health (DOE-HQ) reviewed the methodology used in the safety analysis that was used to set the power levels. SRP'S plan is to develop more accurate methods, improved analysis and data, and to verify and use other criteria in order to raise the power levels. The current review team, who all participated in previous reviews of the SRP safety methodology, were asked to review the duPont action plan on April 21, 1987.

The scope of the review of the action plan encompasses the plan objectives, proposed methodology for accomplishing objectives, plan organization, schedule and resource credibility, evidence of planning depth, evidence of management commitment to the plan, assessment of requisite technical depth and understanding of the technical issues. In the several days available for this initial review, the team's evaluation is confined to a top level of scrutiny. We believe this is adequate to assess the action plan presented.

The methodology used by the team to review the action plan was to have each team member concentrate in the area(s) of his expertize. During evaluation of the action plan each team member, as appropriate, considered the following classes of attributes to determine if the action plan adequately addresses these important attributes: Safety, Operations, Design, Costs, Regulatory Requirements, and Administration.

There are major subdivisions that one can consider within each class. However, it was not necessary to delve into that level of detail for the summary action plan presented.

1.1 Organization

The subsequent portion of this report has been divided into two major sections. Overall comments on the action plan are presented in Section 2.0 and specific technical comments are presented in Section 3.0. In each section, the comments are referenced to the relevant section of the action plan as appropriate.

2.0 Overall Comments

2.1 The need that duPont feels to increase the power level of the reactor as expediously as possible is understood by the team. However, the team believes that the effort to resolve the concerns about the ECCS and related issues is being diluted by mixing in the urge to increase power with the safety concerns. In other words, the concern about the performance of the ECCS and related safety issues is the cause of the power reduction. Resolve the cause of the problem and the symptom, power reduction, will be resolved also.

2.2 The problems or concerns to be resolved by the action plan are not clearly and explicitly stated. We suggest that the authors of the action plan clearly and explicitly enumerate the concerns to be resolved and confirm with the groups expressing them that there is agreement on the concerns to be resolved. This will help to minimize misunderstandings and open-loop research, focus the issues to be resolved, and aid in confirming when resolution is accomplished.

2.3 The overall and supporting objectives are not clearly stated. We suggest that the authors of the action plan concentrate on defining and stating these such objectives clearly and succinctly. The credibility of the Performance Requirements and technical specifications for the resolution of the concerns expressed about the ECCS and related issues will follow directly from how well the objectives are formulated.

2.4 The action plan is really not a plan so much as it is a list of tasks to be accomplished in order to achieve higher reactor power. For example, the action plan is a start but it is not specified adequately as to performance requirements, and related technical specifications. Further, the information requirements for and interrelationships between the various supporting analytical and experimental tasks is absent.

2.5 The "PROGRAM ELEMENTS" described in the APPENDIX to the action plan are an initial step. However, the review team believes that, in general, it is not possible to assess the credibility of the program elements presented because:

-times for critical tasks such as element 4.2, are underestimated;

-material resources essential to accomplishing a task such as computer dollars for element 4.3 are absent--they may be quite substantial;

-consistency between program element definition varies;

-responsibility for program element performance appears to be assigned to several individuals which, if true, would not be proper;

-the sequencing between some program elements does not seem logical. For example, see program elements 3.1 and 4.2. Normally, one would select models to be validated before designing and conducting experiments. Perhaps there is an explanation for such sequencing. If so, it would be good to state it. Unless a full demonstration test on the SRP system is contemplated, it is crucial that the analysis and experimental efforts be closely integrated in order for the plan to be credible.

2.6 It would be beneficial in evaluating the overall credibility of the action plan if a statement on the overall scope, manpower, and resources required are given.

2.7 The review team believes it is essential that a comprehensive independent safety assessment take place. This means a separate group of analysts should be used to literally conduct the important analyses and independently verify the duPont work.

2.8 A comprehensive, integrated system transient analysis should be developed. The current lack of such an integrated capability inhibits the ability to properly identify and and assess the importance of interrelated physical phenomena.

2.9 In summary, the action plan is recognized as a reasonable start in the short time in which it was developed. At this point , the action plan is not yet well defined or integrated. If used in its present state, the objective duPont desires to achieve may be accomplished inefficiently or, worse yet, not at all. We suggest that use of some personnel experienced in resolving the subject concerns in the further development of the action plan will be beneficial.

2.10 The review team was asked to comment on whether the action plan was suitable for presentation to the Advisory Committee on Reactor Safeguards (ACRS). Of course, we cannot presume to speak for ACRS. But, based on our experience in making presentations before ACRS and in acting as consultants to ACRS, we suggest that the action plan needs improvement as suggested above before a presentation to ACRS would be worthwhile.

3.0 Technical Comments

3.1 General Technical Comments

In general, the program elements are useful in working toward the evaluation of the LOCA accident and reducing its effects. The system improvements that are proposed are the most useful in that they could produce improvements in safety margins whether or not the margins are now acceptable.

There is not enough detail in many of the program elements to assess whether they are well directed toward their objective. Some of our detailed comments address desirable aspects of the program elements that are not explicitly described.

It is not apparent from the plan how the Best Estimate procedure for safety analysis as opposed to the Conservative approach is to be implemented. The plan seems to contain elements of both approaches and we believe it is necessary to lay out the philosophy of the approach to be used before a comprehensive supporting plan can be formulated.

The plan addresses the issue of ECCS Performance Criteria when the limits are based on core damage, but it is our opinion that the current criteria of no bulk boiling also requires added quantification. Specifically, the permissible fuel/target temperatures that are acceptable under conditions of coolant maldistribution require specification.

From an overall viewpoint, the plan makes several references to single phase conditions during the LOCA transients. While the coolant remains in a liquid state below the boiling point, it is completely misleading to assume that this no boiling criteria "..approach simplifies considerably the transient analysis..". During the majority of the LOCA transient, the reactor system is composed of some quantity of air mixed in some manner with the coolant. This multi-component composition of the cooling fluid is in most aspects at least as difficult a problem to analyze as are the steam-liquid water conditions in commercial power plants. MultiPcomponent coolant composed of non-condensable air mixed with liquid heavy water presents a high degree of difficulty in transient fluid flow and heat transfer analysis.

The plan contains several elements that address the early stages of the accident, the flow instability phase. This was not discussed in detail in the last review. The effect of system variables on flow instability and recovery should be detailed in support of the elements.

In certain cases, it may be better to consider doing subscale experiments rather than or in addition to detailed analysis. A particular case that illustrates the point is the distribution of levels and associated assembly flow rates in the upper plenum. Sub scale experiments could be used to assess the levels and flows under prototypical conditions, including assumed break geometries. It is quite possible that such tests could be performed in a shorter time, at less cost, and with greater resultant credibility than for detailed multidimensional calculations. In this way transient conditions could be addressed as well as the steady state flows. In any event such tests would provide more comprehensive assessment of the analytical methods.

A key point is that perhaps the most important factor as far as enhanced credibility of the safety analysis, that is an independent safety assessment effort by personnel skilled in LWR safety analysis, is not addressed in the reference plan.

3.2 Technical Comments-subtasks

Detailed comments on program elements, listed by program element number follow. Only those elements on which we have comments are listed.

1.1 Performance Objective of ECS for Hypothetical LOCA

This is desirable. Outside resources should include a representative from NRC.

2.2 Annular Flow Distribution Measurements

The purpose is to improve the understanding of low annular flow distributions for heat transfer to the coolant. These are unheated tests and while certainly needed, the additional aspect of high void heat transfer is also needed.

This element tests two hypotheses, tank level and spider orientation, for maldistribution. These tasks are complete, and if a thorough understanding of the sources and the error bands on the flow distribution has not been achieved, more work should be done. The circumferential and temporal distributions should be determined. The nature of the data scatter, whether it is statistical or phenomenological should be determined. This is necessary to resolve whether a statistical bound or a phenomenological lower limit for flow must be used.

2.3 Decay Heat Power Analysis

See proposed NRC regulations for methods on treating uncertainties.

2.4 Review of Experimental Bases for FLOOD

The stated benefit of "Confirmation of steady-state results.." is not necessarily compatible with the goal of obtaining "..minimum ECCS flows..". Transient conditions may result in zero flow on a temporary basis.

2.5 Polybor Temperature Effects on Limits

If the polybor temperature represents a major determinant of reactor power, we would like to understand more about it.

Item 3. Assembly Flow and Heat Transfer

The "enhanced understanding" of heat transfer and hydraulic characteristics should also include the effects of multi-component coolant in addition to bulk boiling. Assembly heat transfer with air water mixtures during the reflood phase may not be amenable to the use of existing correlations. The program should include verification of the heat transfer correlations used during the reflood phase. If there is not sufficient data presently to assure that the correlations are accurate, then heat transfer tests should be performed with heated assemblies and accurate simulation of the air/water/steam conditions in the reflood phase of the LOCA

3.1.1 Annular Coolant Flow Maldistribution

See comment on element 2.2. The element is essential to assessing the safe power regardless of the other elements. It is presumed that this is the element that contains the Steam Induced Diversion of ECCS that concerned the NAS panel. Solution of that concern deserves more description.

3.1.2 Assembly Component Changes

This is desirable both for ECCS and for flow instability, and may be useful for other limits that impact pressure boundary integrity. Without a statement of the ECCS modifications, we can not comment on the specific approach.

3.2 Flow Instability and Recovery Experimental Studies

If there is serious uncertainty about the mechanism of flow instability, this is an important element.

3.3 Analysis of L Area Tests

What is this element?

3.4 Plenum Distribution Measurements

The fluid behavior in the upper plenum is obviously very complex due to the mass of internal hardware and the unique geometry of the plenum. Most of duPont's analyses have been based on water distributions within the upper plenum that were determined in experiments somewhat similar to the conditions to be expected during a LOCA, but the data is for steady equilibrium conditions that probably would not be present during the early time periods (tens of seconds) of an LOCA. It is not unreasonable to imagine that the water in the plenum would be distributed in a somewhat chaotic manner of splashing and surging around the plenum thus creating a condition of temporarily uncovering various assemblies. The total inventory of water in the plenum is such that the central region only has two inches of coolant covering the entrance ports of the hot assemblies on an equilibrium basis. It is easy to speculate that the depth could vary significantly in transient conditions.

We believe that it is important to understand transient effects in the plenum and their impact on predictions derived from steady state tests. It should look at gas and vapor deentrainment, at "sloshing" in the plenum, and at channel vapor flow and pressure transients. The determination of transient flow distributions in the plenum are very important and experimental data for this is deemed necessary.

3.5 Pump and Heat Exchanger Performance During LOCA

The task should also include any information needed to define the flow rates through cavitating pumps and system components. This will require careful evaluation of the system codes that are used.

4. Computational Assessment of Severe LOCA

The statement "The calculations will be normalized to the experimental data .." is an oversimplification. Calculations should be qualified and verified with experimental data. If significant disagreement exists, the cause should be determined and corrected.

4.1 Determine Best-Estimate Computer Codes

The computer codes should handle the non-condensable mixture of air and water. This technical aspect is available in several codes, but it should be recognized that the incorporation and verification/qualification of non-condensables in these codes has never received major attention.

4.2 Determine and Implements Model Improvements

The models for flow instability will need particular emphases.

4.3 Verify and Validate Computer Codes and Models

Here technical detail is obviously missing.

4.4 Perform Best-Estimate Calculations for SRP Reactors

This element should include the sensitivities necessary to derive bounding cases from the best estimate calculations.

4.5 Validate and Peer Review Results of Calculations

Confirmatory analyses should be performed as a part of the objective of ensuring accuracy.

5.0 Credibility of Coolant System Breaks in SRP Reactors

This total element, "Credibility of Coolant System Breaks in SRP Reactors" , is more than that. It seems to be seeking a credible break scenario and its thermal hydraulic consequences. This seems primarily to be aimed at the early, flow instability phase of the LOCA transient, and only ruling out large pipe breaks would seem to affect the ECCS part of the transient. If this is a primary objective, however, it is not well described.

This is a valid area of study. It should be remembered that commercial reactors have expended much study on these problems. The low pressure and the low stored PV energy in the system may make this effort more meaningful for SRP than for high pressure systems. The consultants listed seem well suited to the problem. However, they may not be expert on some of the concerns listed below.

The comments we previously have made about sensitivities and the need to understand them are particularly valid here, where rapid, transient flow and pressure effects are important. The effect of small variations may have a large effect in the course of a flow instability event. The dynamics of crack opening and local flow and pressure effects that interact with the opening are important in determining the system effects and are likely to have a large error band. The error band must be bounded to validate the predictions.

If such rapid failures are used as a basis for an accident analysis, the system codes may not simulate well the system behavior. Waterhammer effects should be considered if this mechanistic approach is followed. Dynamic loads on components such as the plenum should be evaluated. Pipe whip effects should be evaluated. Specific comments on the elements are as follows:

5.1 Dynamics of a Double Ended Pipe Break

We interpret this as trying to derive flow and pressure input to a LOCA from a complete shear. We presume that a part of this is to determine the likelihood of such an event; the major element title suggests it, but none of this element describes it. Inasmuch as this event is what commercial reactors assume, prima facie, this should be addressed directly. Other effects of a double ended break described above should be considered.

5.2 Dynamics and probability of a Hinged Pipe Break

See above.

5.3 Pipe inspection and Repair

This seems to be the existing program except for the item. "Characterize potential failure location ...." With respect to the present program, previous recommendations about use of Section 11 of the ASME Code apply, and we presume that Mr. Bush will assist in assuring this.

With respect to failure location, certainly there are different probabilities in the 5 psi portion of the piping than in the high pressure side, and other factors apply. If the LOCA approach is to be abandoned in favor of a specified location, it is inappropriate to assume only the most likely location, and pipe breaks that are representative of a full spectrum of breaks above some minimum probability must be represented in the analysis. The objectives thus must define this minimum probability.

6.1 Install Fourth ECS Addition System in L, P, and K Areas

Installation of additional ECCS capacity should have high priority inasmuch as it has the clearest probability of successfully resolving the concerns.

6.2 Development of Leak Flow Restrictor

This is unclear; is the objective to reduce initial break flows, or to hold ECCS water in the plenum. The latter objective seems more feasible and useful. The added risk potential is also obscure. This is hard to judge without a concept.

The impact of these type of design changes on accident scenarios other than DEB LOCA should be evaluated. If the design change is a flow restriction, then high pipe velocities, and consequent erosion, choking, and flow transients and their effect on other accidents, system integrity and dynamic loads is a concern. If it is a check valve or similar device, it is capable of failures and waterhammer events that are new accident initiators. If it is a weir or similar dam for ECCS water the flow restrictor comments apply.

7.3 Expert Review of ECS Program

As mentioned above, detailed peer reviews and validating analyses are desirable besides the expert review.

7.4 Revise Procedures and Training for ECS System Modifications

Besides the training on the fourth addition system, a detailed review of the procedures with respect to system interactions and effects of other operator responses is warranted.

7.5 Update Predictive and Preventive Maintenance Program for ECS Equipment

The relation of this item to the program is unclear.

DPST-87-1000

AN ACTION PLAN TO DEMONSTRATE THE ADEQUACY OF THE SRP REACTOR SYSTEM RESPONSE TO A HYPOTHETICAL SEVERE LOSS-OF-COOLANT ACCIDENT

M. R. Buckner
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R. J. Pryor
C. P. Ross
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EXECUTIVE SUMMARY

The Savannah River Laboratory (SRL) has initiated an aggressive program of improvement activities to extend its capability to analyze the response of the Savannah River Plant (SRP) reactors to a hypothetical, worst case loss-of-coolant-accident (LOCA). SRP has also initiated an aggressive program to modify the reactor systems to improve their response to a severe LOCA. The goal of these programs is to significantly strengthen the technical basis for safe operation of the SRP reactors at full power levels.

The programs entail improvements on four parallel, but interrelated,fronts:

- Development of a set of improved calculational tools for establishing the technical basis for safe operation of SRP reactors given the reactor design and the postulated LOCA conditions.
- Enhancements to the experimental data base to support analysis of postulated LOCA conditions and to benchmark the computer codes used in the analysis.
- Establishment of a more physically meaningful and credible technical definition of the postulated severe LOCA than the hypothetical, instantaneous double-ended guillotine break now imposed on the analysis.
- Modifications to the emergency cooling systems (fourth emergency cooling addition system), to the reactor hydraulic system and to the fuel/target assemblies to improve the reactors' ability to respond to the postulated LOCA.

Detailed tasks and their interrelationships are defined in the Action Plan for each area of improvement.

Since SRL and SRP are moving expeditiously on four parallel fronts, improvements made in one area can obviate the need for, or importance of, improvements in another area. The program of interrelated activities provides the programmatic options necessary to accomplish the dual goals of protecting the public health and safety while establishing appropriate power levels.

Figure 1 indicates the key anticipated decision points corresponding to major improvement milestones. As these decision points are reached, the improved technical basis and proposed new power limits will be independently reviewed and approved prior to increases in reactor power.

The projected milestones shown in Figure 1 include:

- Improvements in the calculational tools and experimental database to establish and validate the technical basis for setting power. Limits will be developed initially for conditions that allow no bulk boiling (NBB-1, NBB-2) and subsequently for conditions that assure a maximum cladding temperature is not exceeded that would impair core cooling (MAXST-1) These major milestones are discussed in detail in the Action Plan.

- Installation of a 4th ECS addition system at L reactor and subsequent installation at P and K reactors.

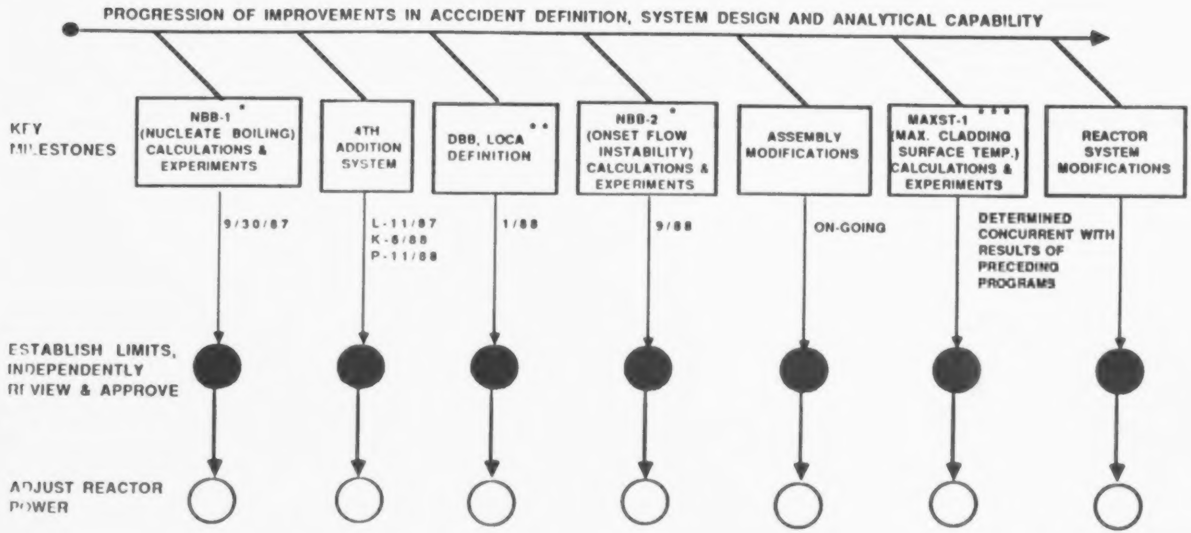
- Establishment of detect before break program (DBB) and re-definition of the postulated initiating LOCA.

- Additional modifications to the reactor complex and fuel/target assembly designs.

The tasks in the Action Plan have been divided into a few major time phases associated with the planned improvements in the calculational and experimental models. Tasks associated with the near term phases have been defined, planned, and scheduled in detail. As these tasks are completed, the quality of the information necessary to define, plan, and schedule the improvements included in the later phases will increase and will be used to assess the appropriateness of currently planned improvements. A summary of the interrelationships of these tasks is shown in Figure 2. Subsequent revisions to this Plan will reflect the information generated and decisions made during the earlier phases.

FIGURE 1

OVERALL ACTION AND DECISION PLAN


PROGRESSION OF IMPROVEMENTS IN ACCCIDENT DEFINITION, SYSTEM DESIGN AND ANALYTICAL CAPABILITY
KEY MILESTONES
NBB-1 * (NUCLEATE BOILING) CALCULATIONS & EXPERIMENTS
4TH ADDITION SYSTEM
DBB, LOCA DEFINITION **
NBB-2 * (ONSET FLOW INSTABILITY) CALCULATIONS & EXPERIMENTS
ASSEMBLY MODIFICATIONS
MAXST-1 *** (MAX. CLADDING SURFACE TEMP.) CALCULATIONS & EXPERIMENTS
REACTOR SYSTEM MODIFICATIONS
9/30/87
L-11/87
K-6/88
P-11/88
1/88
9/88
ON-GOING
DETERMINED CONCURRENT WITH RESULTS OF PRECEDING PROGRAMS
ESTABLISH LIMITS, INDEPENDENTLY REVIEW & APPROVE
ADJUST REACTOR POWER

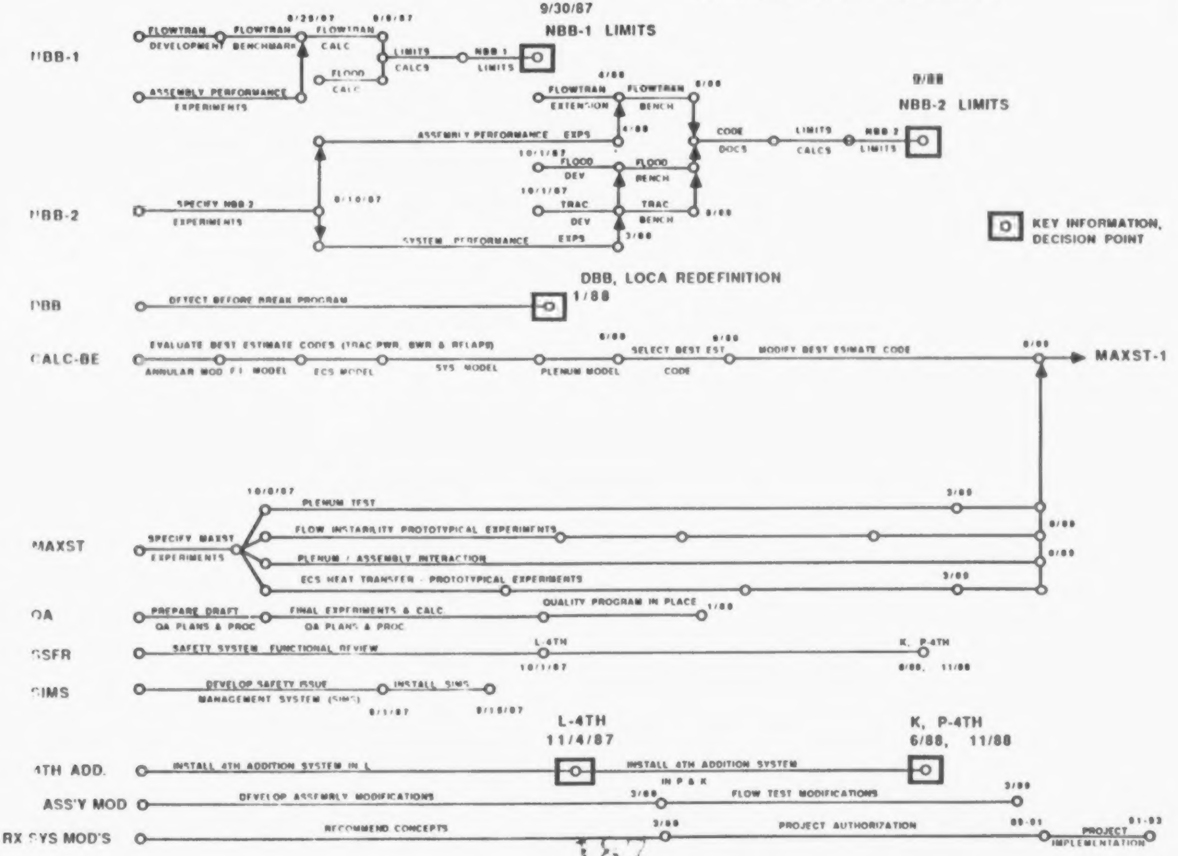
NOTE: ALTHOUGH SHOWN AS A TIME LINE FOR SIMPLICITY, PROGRAMS ARE PROGRESSING CONCURRENTLY

KEY:

*	NO BULK BOILING (NBB)
**	DETECT BEFORE BREAK (DBB)
***	MAXIMUM SURFACE TEMPERATURE (MAXST)

FIGURE 2

ECS ACTION PLAN SUMMARY SCHEDULE


9/30/87
NBB-1 LIMITS
NBB-1
FLOWTRAN DEVELOPMENT
FLOWTRAN BENCHMARK
8/28/87
FLOWTRAN CALC
9/8/87
FLOOD CALC
LIMITS CALCS
NBB 1 LIMITS
ASSEMBLY PERFORMANCE EXPERIMENTS
4/88
FLOWTRAN EXTENSION
FLOWTRAN BENCH
6/88
9/88
NBB-2 LIMITS
ASSEMBLY PERFORMANCE EXPS
4/88
CODE DOCS
LIMITS CALCS
NBB 2 LIMITS
10/1/87
FLOOD DEV
FLOOD BENCH
10/1/87
TRAC DEV
TRAC BENCH
6/88
NBB-2
SPECIFY NBB 2 EXPERIMENTS
9/10/87
SYSTEM PERFORMANCE EXPS
3/88
KEY INFORMATION, DECISION POINT
DBB, LOCA REDEFINITION
1/88
DBB
DETECT BEFORE BREAK PROGRAM
CALC-BE
EVALUATE BEST ESTIMATE CODES (TRAC PWR, BWR & RELAP5)
ANNULAR MOD
F.I. MODEL
ECS MODEL
SYS MODEL
PLENUM MODEL
6/88
SELECT BEST EST CODE
9/88
MODIFY BEST ESIMATE CODE
9/88
MAXST-1
MAXST
SPECIFY MAXST EXPERIMENTS
10/8/87
PLENUM TEST
3/88
FLOW INSTABILITY PROTOTYPICAL EXPERIMENTS
9/88
PLENUM / ASSEMBLY INTERACTION
9/88
ECS HEAT TRANSFER - PROTOTYPICAL EXPERIMENTS
3/88
QA
PREPARE DRAFT QA PLANS & PROC
FINAL EXPERIMENTS & CALC QA PLANS & PROC
QUALITY PROGRAM IN PLACE
1/88
SSFR
SAFETY SYSTEM FUNCTIONAL REVIEW
L-4TH
10/1/87
K, P-4TH
6/88, 11/88
SIMS
DEVELOP SAFETY ISSUE MANAGEMENT SYSTEM (SIMS)
8/1/87
INSTALL SIMS
9/15/87
L-4TH
11/4/87
K, P-4TH
6/88, 11/88
4TH ADD.
INSTALL 4TH ADDITION SYSTEM IN L
INSTALL 4TH ADDITION SYSTEM IN P & K
ASS'Y MOD
DEVELOP ASSEMBLY MODIFICATIONS
3/88
FLOW TEST MODIFICATIONS
3/89
RX SYS MOD'S
RECOMMEND CONCEPTS
3/89
PROJECT AUTHORIZATION
89-91
PROJECT IMPLEMENTATION
91-93

Appendix D

Aging: Technical Discussion

The Savannah River reactors were constructed in the 1950s. N Reactor began operation in 1963. The committee has been unable to ascertain the lifetimes projected for these reactors when they were designed, but conventional practice [1] would suggest they were designed for lifetimes of 20 to 40 years. It is evident, then, that the reactors are approaching the end of their original design lifetimes. Both the Savannah River reactors and N Reactor are now encountering material aging phenomena that pose problems for the long-term operation of these reactors.

STRESS-CORROSION CRACKING AT THE SAVANNAH RIVER REACTORS

The primary system boundaries of the Savannah River reactor systems are constructed of type 304 stainless steel, which by today's standards contains relatively significant amounts of carbon—300 to 800 ppm. When the power of the Savannah River reactors was increased in the 1960s, it was found necessary to acidify the D_2O moderator and primary coolant with nitric acid (deuterium ion concentration $= 10^{-5}$ molar) in order to inhibit corrosion of the aluminum fuel cladding and to avoid contamination difficulties caused by corrosion products suspended in the D_2O [2].

To prevent radiolytic decomposition of nitrate ions, which would cause attack on the aluminum fuel cladding, it was found necessary to operate the reactors with 0.5 to 2 ppm oxygen dissolved in the D_2O [3]. This dissolved oxygen augments the oxygenation provided by 1-3 ppm deuterium peroxide formed in the D_2O by the radiolytic reactions

$$D_2O \rightarrow D + OD$$

$$2\,OD \rightarrow D_2O_2.$$

These high oxygen concentrations contrast with oxygen concentrations of about 20 ppb sought in commercial boiling water reactors [4]. The high oxygen concentrations make sensitized regions of the high carbon-content stainless steel susceptible to oxygen-induced, intergranular, stress-corrosion cracking [5]. Welding is the most common mechanism for sensitizing stainless steel to stress corrosion cracking, and the Savannah River reactor piping systems and vessels employ welded constructions.

Cracking, believed to be intergranular stress corrosion cracking, was detected in the so-called "knuckle" region near the base of C Reactor (see Figure D.1) in 1967 [6]. Evidence of stress corrosion cracking has been observed in nozzles of the R and L reactors and an effluent line in C Reactor [6]. A recent inspection of 794 welds in the piping systems of the Savannah River reactors identified 53 instances of cracking believed to be caused by stress corrosion. Cracking of the reactor system becomes a major safety hazard only if the cracks or flaws grow sufficiently that they propagate rapidly under the loads of normal or abnormal operations and cause catastrophic failure of the system. In this regard, the possibility of catastrophic failure of a reactor vessel is the principal concern.

Cracking in the piping system may be corrected—and in 11 instances is being corrected—by replacing the affected pipes. Replacing the reactor vessels, however, has been deemed infeasible. Repair of cracks in the reactor vessels is proving to be a technological challenge. In 1968 patches were welded over cracks in the knuckle region of C Reactor [7]. In 1984, leakage in the heat affected zones of these welds was detected. The pinhole leaks found at the perimeters of the patches were attributed not to stress corrosion cracking but to the accumulation of helium bubbles that

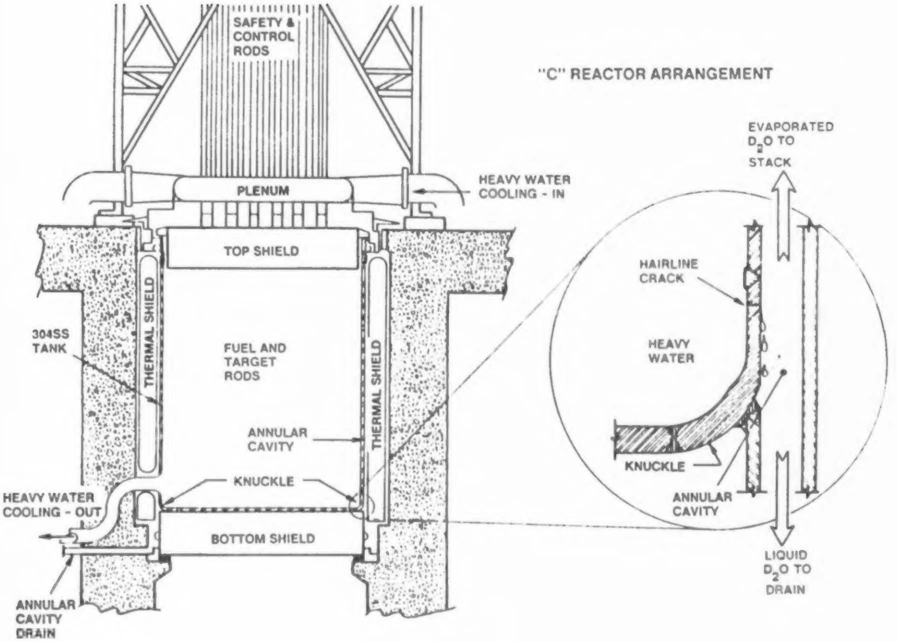

SAFETY & CONTROL RODS
"C" REACTOR ARRANGEMENT
EVAPORATED D_2O TO STACK
PLENUM
HEAVY WATER COOLING - IN
TOP SHIELD
HAIRLINE CRACK
HEAVY WATER
304SS TANK
THERMAL SHIELD
FUEL AND TARGET RODS
THERMAL SHIELD
ANNULAR CAVITY
KNUCKLE
KNUCKLE
ANNULAR CAVITY
HEAVY WATER COOLING - OUT
BOTTOM SHIELD
LIQUID D_2O TO DRAIN
ANNULAR CAVITY DRAIN

FIGURE D.1 The "knuckle" region of the C Reactor and leaks caused by oxygen-induced, intergranular, stress-corrosion cracking.

had formed an interconnected porosity at grain boundaries in the heat affected zones of the welds [8,9]. The helium is formed by the following sequence of nuclear reactions:

$$^{58}\text{Ni} + \text{n} \rightarrow {}^{59}\text{Ni}$$

$$^{59}\text{Ni} + \text{n} \rightarrow {}^{56}\text{Fe} + \ .$$

In the knuckle region, the helium accumulation is thought to be less than 10 atomic ppm [10]. In other regions of the reactor vessels at Savannah River, accumulations in excess of several hundred parts per million are expected. Welding in these areas of high helium concentrations could be expected to be more prone to leakage than in the knuckle region.

TREATMENT OF THE STRESS-CORROSION CRACKING PROBLEM AT SAVANNAH RIVER

Stress-corrosion cracks have caused C Reactor to be removed from service, apparently permanently. The extensively cracked knuckle region is unique to C Reactor, and there are a number of reasons to believe this region was particularly prone to stress-corrosion cracking [11]. Nevertheless, there is no reason to believe P, K, and L reactors are immune to stress corrosion particularly in the regions of high-energy welds. Unfortunately, there are insufficient data yet available to make meaningful predictions of when stress-corrosion cracking might develop in P, K, and L reactors. In any event, short of radical alterations in the materials and water chemistry used in the Savannah River reactors, there is no way to prevent the conditions of stress-corrosion cracking from arising. Helium embrittlement makes conventional welding unsatisfactory as a method for repairing any cracks that do develop. The Savannah River contractor has chosen to confront the issue of stress-corrosion cracking with a combination of "detect-before-fracture" and "leak-before-break" philosophies [8,12]. That is, because cracks must reach a critical size before they propagate unstably and cause catastrophic failure, there should be a high probability that cracks and leaks will be detected sufficiently early to permit safe shutdown of the reactors.

The Savannah River contractor has introduced a broad range of programs to implement these philosophies [8,13]. These programs have benefitted from extensive use of consultants and expertise throughout the country. One of the most important of these programs is enhanced surveillance of vessel welds in P, K, and L reactors [14]. K and L reactors have been examined and the inspection of P Reactor will soon be completed. At present, only visual inspections of crack indications with dye-penetrant confirmation is available through surveillance. Such visual procedures are not universally accepted as reliable means for detecting cracks. The Savannah River contractor is developing the capability to employ ultrasonic techniques and is evaluating the potential of eddy-current methods [8]. The committee believes efforts to improve the technology for detecting cracks in the Savannah River reactors should be encouraged.

Once cracks are detected, it remains to be determined if they will propagate under the loads imposed by normal operation or accident conditions. Stress analyses have been done on the L- and C-Reactor vessels [11,15]. The analysis for L Reactor used conventional fracture mechanics analyses within the constraints specified by the American Society of Mechanical Engineers (ASME) Pressure Vessel Code [16]. The decidedly less conservative analysis of cracking in C Reactor [11] would be judged less acceptable by a broad spectrum of the technical community, but because of the C-Reactor retirement the analysis will not be redone.

The L-Reactor analysis should be taken as only illustrative of a technology to be applied once cracks have been detected. The analysis considered loads arising in a normally operating reactor during a design basis earthquake [17] and under hypothetical accident conditions. The accident conditions analyzed were very severe. Fracture mechanics analyses were performed for a hypothesized axial crack at the mid-level of the vessel and a crack in the vicinity of the outlet nozzles. Presumably, the ongoing probabilistic risk assessment of the Savannah River reactors will provide a more complete understanding of possible loads on the reactor systems and crack locations and orientations.

Data on the fracture toughness of the reactor vessel steel are critical input to the fracture mechanics analyses. Currently, the effects of radiation hardening of the steel [18,19] are assumed to reduce the fracture toughness to 1/10 its unirradiated value, which is itself conservatively estimated to be 2000 in-lb/in^2. Savannah River has launched a program investigating irradiated materials properties to confirm that these input data are in fact conservative [20]. Data have been obtained from neutron fluences approximately equivalent to that sustained by the most irradiated portions of the vessels. However, shutdown of the High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory has prevented obtaining data for higher fluences, so the available data may only be used to confirm conservatism.

Conservatisms, which may seem of an extreme nature, are warranted in the implementation of leak-before-break and detect-before-fracture procedures because of the many uncertainties. These conservatisms must be highly probable of compensating for undetected or ill-characterized flaws, cracks that do not leak, unanticipated loads, and uncertain material properties.

Finally, Savannah River has also introduced a program to evaluate alternatives to conventional welding for repair of cracks in the vessel [21]. Though none of the alternative technologies has yet been shown to be completely applicable, the committee believes this investigation of repair methods should be sustained at Savannah River and fully supported by DOE.

GRAPHITE GROWTH AND PROCESS TUBE EMBRITTLEMENT AT N REACTOR

The acute aging problems at N Reactor involve materials within the reactor core. The graphite moderator block has sustained high, but unequal, neutron fluences over the years. As would be expected [23], radiation damage to the graphite initially caused the moderator block to shrink. Now, however, the regions of highest fluences in the moderator are beginning to expand (see Figure D.2). The varying dimensions of the moderator have caused some damage to the block and have distorted horizontal control rod channels and vertical channels for the backup scram system. Concerns have developed that offsets between graphite stacks will place stresses on the process tubes and the graphite cooling tubes. Expansion of the graphite moderator has been sufficient that in the near future it will thrust against the top shield of the reactor vault. When this happens, continued expansion of the graphite will be absorbed by void spaces in the core. The threat of unacceptable stresses on tubing and distortion of channels will then become severe. Methods to enhance in-service inspection of the interstitial regions of the graphite stack need to be developed to monitor unexpected graphite distortion and damage, and to provide early warning of pending tube distortions.

Process tubes used in the N-Reactor core are being embrittled by two principal mechanisms:

- Radiation damage, and
- Zirconium hydride formation.

As a result, the tubes are becoming more susceptible to fracture at a time when stresses on the tubes are expected to increase. The rupture of a single process tube is thought not to pose a significant safety hazard [24]. But, as embrittlement progresses, there is an

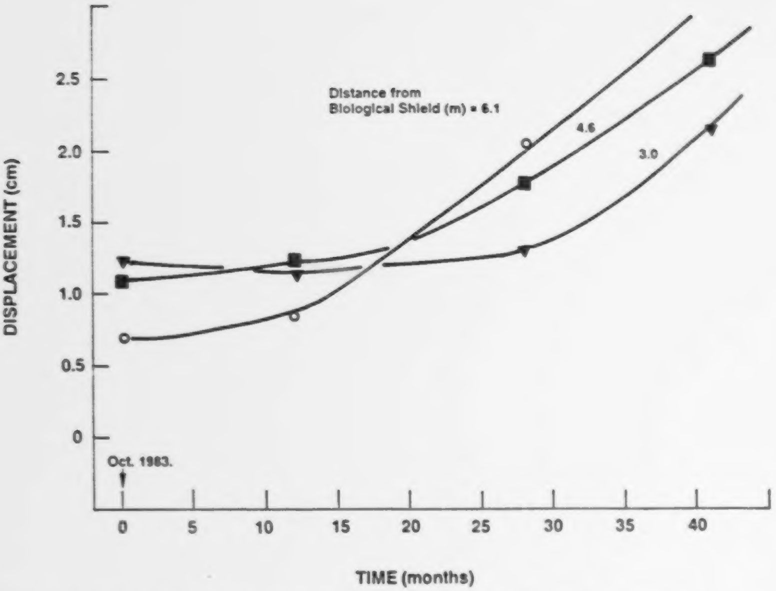


FIGURE D.2 Expansion of graphite moderator block at N Reactor.

increasing danger that fracturing of a pressurized process tube will propagate to adjacent tubes, and in turn lead to a severe core damage accident [25a,b].

TREATMENT OF AGING PROBLEMS AT N REACTOR

Graphite growth will soon end the useful life of N Reactor. The Assistant Secretary for Defense Programs has indicated it is DOE's intention to retire N Reactor from service once expansion brings the graphite moderator into contact with the upper shield of the reactor vault [26]. The contractor projects that this will occur sometime between 1991 and 1996 [27,28]. In the meantime, the contractor has plans to slow the rate of growth of the graphite by more uniformly distributing the neutron flux over the moderator volume [28]. The contractor indicates this modification will have the additional benefit of improving the quality of nuclear material produced by N Reactor. A difficulty to be overcome before implementing this mitigation effort is to assure that the revised core configuration is not susceptible to xenon instabilities.

In addition, a program of surveillance and modeling has been initiated [28]. Presumably, these activities will indicate whether other life-limiting conditions are imminent, such as:

- Deformation of the 640 graphite and shield cooling system tubes or
- Deformation of the 84 horizontal control rod channels.

Unfortunately, materials data needed for the modeling efforts may not be available as planned since HFIR has been shut down.

The contractor operating N Reactor has suggested a variety of options for correcting the problems arising as a result of graphite expansion [27, 29]:

- Trimming the top and bottom of the graphite moderator block,
- Alternate control rod designs to mitigate control channel curvature effects,
- Trimming graphite to provide clearance for ball-hopper collars,
- Protective measures to preserve integrity of the graphite and shield cooling tubes, and
- Removal of "kerfs" from the untubed channels at the top and the bottom of the graphite stack.

Provided graphite contraction and expansion has not caused unacceptable damage to the moderator block, some of these options appear feasible.

The N-Reactor contractor has implemented the recent suggestions of DOE's Design Review Team [30] to increase the surveillance of process tubes. Initially, the work was planned to involve the destructive examination of four process tubes and the nondestructive examination of 50 process tubes. Nondestructive examinations of tubes by eddy-current techniques and by two types of ultrasonic techniques indicated there were flaws in some tubes. As a result, the nondestructive examinations were expanded to involve 152 tubes, including complete examination of all tubes in the class having the most numerous indications of flaws. Twenty-one indications of flaws were found in 13 of the tubes examined. Recently, comparisons have been made of the flaw sizes estimated by ultrasonic and eddy-current techniques with those found by destructive metallurgical methods. This comparison showed that the nondestructive techniques indicated well both the axial and the radial locations of flaws. However, the nondestructive techniques underestimated the length of flaws and in the case of flaws deeper than 0.1 inch (the thickness of the process tube wall is 0.275 inches) underestimated the depth of the flaw.

The contractor has concluded that the nondestructive examinations are reliable for flaw indication no deeper than 0.1 inch provided that the largest flaw size indications obtained by the various methods is accepted. When flaws are indicated to be deeper than 0.1 inch, the contractor plans to remove the flawed tube from service. The destructive examination of tubes will be expanded. To date, six tubes have been removed for such examinations during the current outage.

The surveillance of the N-Reactor tubes has recently suggested that fret marks (abrasions) on the tubes may initiate cracking. Surveillance of tube fretting and possible crack initiation has been undertaken.Complete retubing of the N Reactor has been deemed by the contractor to be impractical. Only nine replacement tubes are available in inventory. Were adequate funds available, it would require six months to re-establish tube manufacturing capability at the site.

The N-Reactor contractor does not plan to conduct 100 percent nondestructive examination of the process tubes. Yet a defensible statistical analysis of the basis for partial examination of the process tubes has not been prepared. The tubes currently being used in the N Reactor have been manufactured using four different methods. Past studies have shown radiation damage to the tube correlates with both location in the core and method of tube manufacture [31]. In light of this multivariate dependency, the committee believes that a justification of the sampling methods being employed needs to be prepared.

The contractor is studying tube failure propagation and the ability of tubes to withstand bending stresses. Destructive testing of at least one irradiated tube is planned.

An area that deserves but has not received much attention is the effects of shock quenching of the flawed and irradiated process tubes in the event of an accident [33]. The consequences of fracturing multiple tubes during emergency coolant injection have not been included in safety analyses of the N Reactor.

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Appendix E
Effluents: Technical Discussion

In the event of severe reactor accidents at either the Savannah River reactors or N Reactor, radionuclide-contaminated liquids will be discharged from the reactor confinements. At Savannah River [1], the discharge will be first to a 60,000-gallon tank and then to a 500,000-gallon tank, both of which are vented back into the reactor confinement. Overflow from these tanks, however, will be to an open, 50,000,000-gallon basin. At N Reactor [2], discharge is to a 6,300,000-gallon excavated pit or "crib."

Accident mitigation at N Reactor involves the actuation of confinement sprays to sweep radionuclides from the confinement atmosphere [2]. It is inevitable, then, that liquids discharged from the plant in an accident would be heavily contaminated with radionuclides. Peak concentrations of radioactivity are thought to develop in the liquid discharge 2.5 to 10 hours after an accident has begun [3]. Failure to initiate pumping to discharge liquids from N Reactor while maintaining emergency core cooling system (ECCS) and spray system flows could lead to undirected flooding of radionuclide-contaminated liquids from the plant [2].

Contaminated liquid effluents are discharged from the N Reactor during normal plant operation. In the past these discharges have been at a nominal rate of 1330 gallons per minute and have contained about 6000 curies per year of radioisotopes with half-lives greater than 48 hours. Radioactive isotopes of strontium and cobalt have been the predominant contaminants.

The discharge of contaminated liquid effluents from the N Reactor is an environmentally poor practice and has been criticized as such in previous reviews of the reactor [4,5,6]. The contractor for the N Reactor has recently taken steps to reduce the discharge to 505 gallons per minute and about 4100 curies per year. The contractor is currently working on filtration equipment that is expected to further reduce the radioactivity of the discharge to about 1500 curies per year. An ion exchange system, which could be used in conjuction with filtration, has been proposed that is expected to reduce the radioactivity of the discharge to 50 curies per year.

The discharge of liquid effluents is also a violation of the fundamental safety philosophy of confinement adopted at the production reactors and will present a safety hazard in the event of a severe accident. The safety hazard would arise because radionuclides dissolved in the liquid effluents can partition from the aqueous phase back into the atmosphere and thus contribute to the radionuclide release from the plant. This partitioning would easily occur for small amounts of noble gases and tritium dissolved in the effluent. But, more importantly, iodine can partition from the effluent into the atmosphere. Iodine that partitions into the atmosphere from effluents in the open basins would bypass the filters and charcoal beds built into the Savannah River and N-Reactor confinements for the purpose of mitigating iodine releases during accidents. Very often, it is the iodine release that would pose the limiting acceptable site boundary doses for the production reactors [1,2].

Iodine partitioning into the atmosphere occurs because iodine in aqueous solution can assume volatile forms. The chemistry of iodine in aqueous media has received considerable attention within the context of accidents analyzed at light-water reactors [7,8,9,10]. The equilibrium chemistry can be understood in terms of the following reactions:

1. Oxidation of iodide:

$$2\,I^- + 2\,H^+ + (1/2)\,O_2 \Leftrightarrow I_2(aq) + H_2O$$

2. Hydrolysis of aqueous iodine:

$$I_2(aq) + H_2O \Leftrightarrow \text{"HOI"} + H^+ + I^-$$

3. Disproportionation of "HOI"

$$3HOI \Leftrightarrow IO_3^- + 2\,I^- + 3H^+$$

4. Vaporization of I_2:

$$I_2(aq) \Leftrightarrow I_2(gas)$$

There has been considerable speculation about the chemical form that would be assumed by iodine upon release from degrading reactor fuel during accidents at the production reactors [11]. Because of the first of the above reactions, it does not matter whether iodine was first trapped in the water as CsI, HI, or I_2; all subsequent chemistry must be considered. The hydrolysis of aqueous iodine is known to be fast [12] and to produce a neutral species conveniently labeled HOI, although in fact, its precise nature is unknown. Oxidation of HOI to form involatile IO_3^- is a kinetically slow process [10]. The equilibrium appearance of I_2 in solution would make possible iodine vaporization. In older literature HOI, too, was considered volatile [7], but vaporization of HOI is now largely discounted [10]. The extent to which iodine escapes the aqueous phases by I_2 vaporization would depend, of course, on mass transport in the ambient atmosphere.

The effects of the complex chemistry of aqueous iodine are conveniently expressed in terms of a partition coefficient defined as follows:

$$\frac{\text{concentration of iodine} - \text{bearing species in solution}}{\text{concentration of iodine} - \text{bearing species in the gas phase.}}$$

Smaller partition coefficients imply a greater tendency for iodine to vaporize from solution.

Inspection of the chemical processes affecting the speciation of iodine in solution shows that the partition coefficient depends on temperature, total iodine concentration, and the hydrogen ion concentration. Some calculated partition coefficients for various conditions are shown in Figure E.1. High temperatures and low total iodine concentrations produce large partition coefficients. High hydrogen ion concentrations are especially effective in producing small partition coefficients. This latter factor in determining the partition coefficients is quite important for the production reactors. Coolant waters at Savannah River are deliberately acidified with nitric acid to inhibit aluminum corrosion [13]. Chemical additives to maintain low hydrogen ion concentrations are not used in sprays or the emergency cooling waters at N Reactor [2]. Acid spills [14] or interaction of the discharged waters with atmospheric gases would raise the hydrogen ion concentrations.

Because the oxidation of HOI to IO_3^- is a slow process, the partition coefficient is time dependent [15]. Accounting for this time dependence requires consideration of a great deal more chemistry [10,16]. Calculated partition coefficients for particular conditions are shown in Figure E.2. It is evident that values lower than the equilibrium partition coefficients discussed above could arise for substantial periods of time.

More recently, it has been recognized that radiolysis of water by dissolved radionuclides could accentuate the propensity for iodine to partition into the gas phase [9,17]. The detailed chemistry of this process is not yet well understood. Key reactants are hydroxyl radical, solvated electrons, and peroxide produced by water radiolysis [18]:

$$4.25\ H_2O \rightarrow 2.70\ e^- + 0.55H + 2.75\ OH$$

$$+ 0.45\ H_2 + 0.70\ H_2O_2 + 2.80\ H^+ + 0.10\ OH^- .$$

Hydroxyl radicals oxidize iodide to iodine:

$$I^- + OH \rightarrow I^\circ + OH^-$$

while solvated electrons and peroxide reduce iodine to iodide:

$$I_2 + H_2O_2 \rightarrow 2H^+ + 2\ I^- + O_2$$

$$I_2 + 2\ e^- \rightarrow 2I^- .$$

So a displacement in the partition coefficient to lower values rather than unlimited iodine formation occurs. It is known that fragments produced by water radiolysis would reduce IO_3^-, though details of the process are incompletely resolved [19].

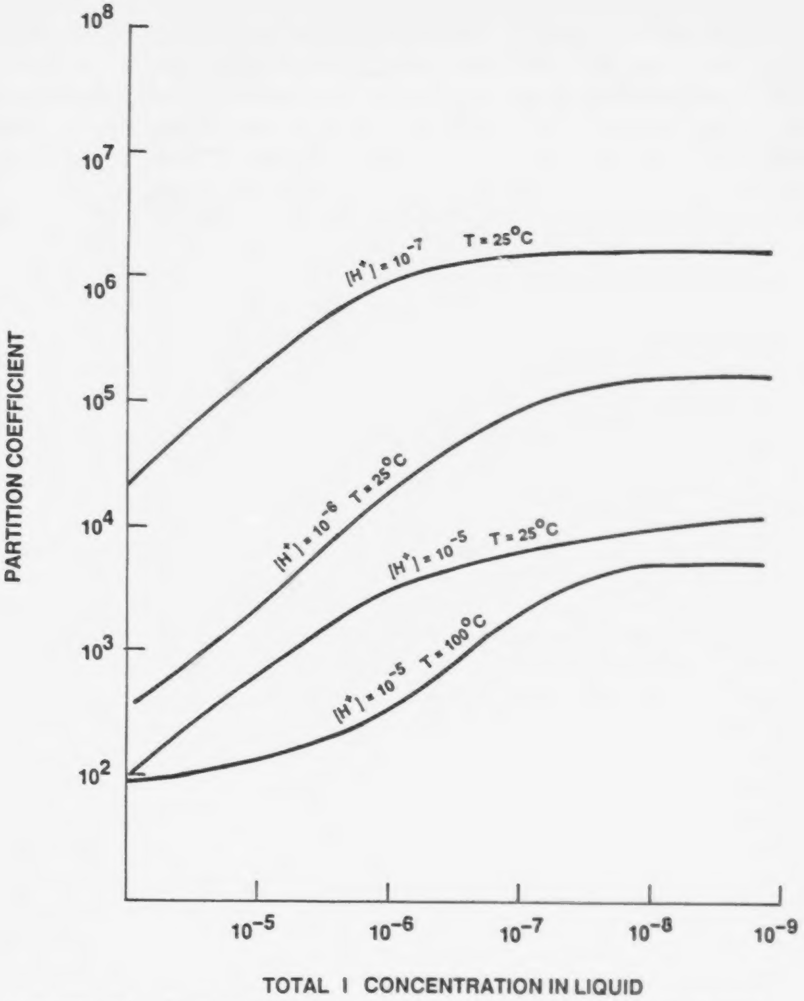

FIGURE E.1 Equilibrium iodine partition coefficient as functions of total iodine concentration in the liquid phase for various hydrogen ion concentrations ($[H^+]$) and temperatures [7].

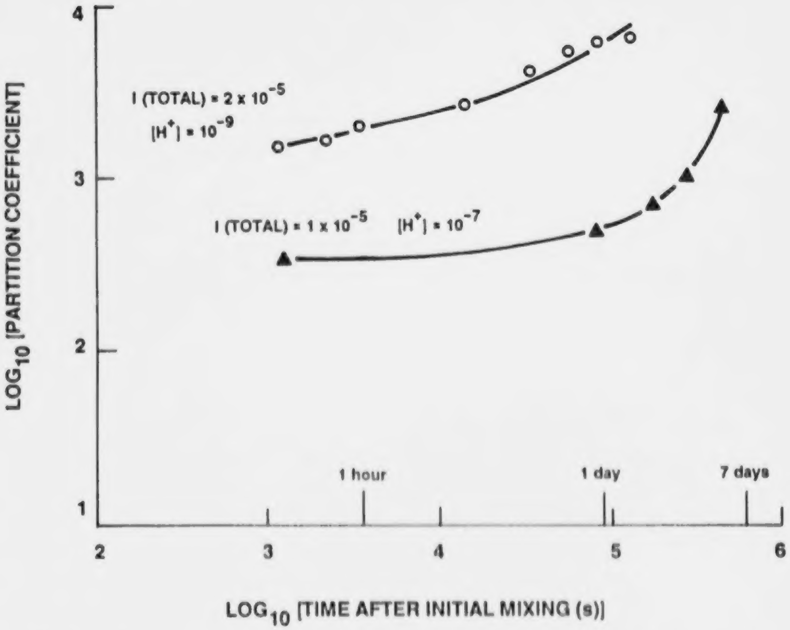

FIGURE E.2 Time dependence of the iodine partitioning coefficient [15].

Radiation effects are not confined to displacement of the partition coefficient. They can also enhance the formation of volatile and poorly water-soluble organic iodides [9]. Beahm and coworkers have shown that methane sparged through irradiated aqueous iodine solutions reacts to form CH_3I to an extent that increases with the hydrogen ion concentration [20]. Further, Beahm et al. [15,21] have shown that aqueous iodine solutions will react under the influence of irradiation with solid organic materials such as polyethylene and ethylene propylene rubber to form methyl iodide.

The radiation induced reactions of aqueous iodine solutions with organic materials may have particular significance for recent proposals to upgrade the cribs at N Reactor. The most recent proposal [22,23] includes lining and covering the crib with a plastic—currently thought to be polyethylene. While the covering would retard but not eliminate partitioning of iodine from the aqueous phase, it would also provide reactants for the formation of volatile organic iodides.

There is no evidence that partitioning of iodine into the atmosphere was considered in analyses of radionuclide releases during hypothetical accidents at the Savannah River reactors or at N Reactor [1,2]. Recently, attention has been given to iodine partitioning in the design of sprays for improved confinement systems at Savannah River, though radiolysis effects on IO_3^- have not been considered [27]. Iodine partitioning is being considered in the redesign of cribs at N Reactor [23].

Iodine may not be the only radionuclide that partitions from the liquid effluent either because of formation of volatile species or reaction to form volatile organic species. Tellurium, selenium [24, 25], and ruthenium [26] may also partition, but the potential partitioning of these species has not been extensively studied.

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Appendix F
Cermet Fuel for SRP Reactors: Technical Discussion

Over the last several years, the Savannah River contractor has been developing an alternative fuel for the reactors. Currently, an alloy consisting of up to 30 weight percent uranium metal in aluminum and clad with aluminum is used as the fuel [1]. Buildup of ^{236}U in the fuel (as the uranium purified of fission products is reused as fuel) will eventually force the use of fuel with higher loadings of uranium [2]. The alternative fuel being developed to meet this need is a cermet composed of U_3O_8 particles dispersed in an aluminium matrix. Because of the method of manufacture [3,4], the fuel is often called "powder metallurgical" fuel. Studies to date indicate that for normal operations, the fuel would be a satisfactory replacement for the existing uranium metal-aluminum fuel [5]. A U_3O_8-Al fuel has been used in other reactors [6].

The committee's concern with the cermet fuel is not related to low-temperature, normal operation of the Savannah River reactors but to accident conditions when the cooling of the fuel might be, at least locally, interrupted. The concern arises because the constituents of the fuel, U_3O_8 and Al, are inherently incompatible. The potential chemical reaction between the fuel constituents is:

$$[(16 - 6x)/3]\ Al + U_3O_8 \rightarrow 3UO_x + [(8 - 3x)/3]\ Al_2O_3,$$

where the stoichiometry of the uranium-bearing product, UO_x, is dictated by the oxygen potential of the reaction product mixture. For most fuel compositions, aluminum will be the excess reactant so the oxygen potential will be given by:

$$2\,\mathrm{Al} + (3/2)\mathrm{O_2} \Leftrightarrow \mathrm{Al_2O_3}.$$

Using the oxygen potential derived for the $\mathrm{Al/Al_2O_3}$ couple, neglecting the slight solid solubility of urania in alumina [7], and using Blackburn's model of the stoichiometry of urania [8], it can be found that the product of reaction will be $\mathrm{UO_2}$ at temperatures less than 1000 K. The product urania becomes hypostoichiometric with increasing temperature so that at 2000 K the uranium bearing product is $\mathrm{UO_{1.9826}}$. At these elevated temperatures, hypostoichiometric urania develops an activity of uranium metal so that eventually the uranium oxide is reduced completely to uranium-aluminum intermetallic compound. Neglecting the slight tendency for hypostoichiometry at elevated temperatures, the initial overall reaction is:

$$(4/3)\,\mathrm{Al} + \mathrm{U_3O_8} \rightarrow 3\,\mathrm{UO_2} + (2/3)\,\mathrm{Al_2O_3}.$$

At 298 K, this exothermic reaction yields about 225 calories of heat per gram of $\mathrm{U_3O_8}$ reacted. The reaction is a metallothermic oxidation-reduction reaction analogous to, but not as exothermic as, the classic thermic reaction [9]. Subsequent reduction of the $\mathrm{UO_2}$ is not especially exothermic. Thermal excursions that might be produced under adiabatic conditions in the cermet fuel by the metallothermic reaction depend on both the initial temperature and the amount of $\mathrm{U_3O_8}$ present in the fuel, as shown in Figure F.1. Were reaction initiated at 373 K, then aluminum melting could be produced in fuel containing more than about 44 weight percent $\mathrm{U_3O_8}$. Ignition of the reaction at the melting point of pure aluminum (about 933 K) could produce temperatures of 1500 to 1750 K in fuel containing 48 to 58 weight percent $\mathrm{U_3O_8}$ as in fuel being tested for the Savannah River reactors [10]. Such high temperatures would be of concern if they accentuated the release of radionuclides or accelerated the rates of steam reaction with aluminum [25]. Cladding on the fuel would reduce the temperature excursions experienced as a result of the reaction.

The cermet fuel being developed at Savannah River undergoes several high-temperature steps during processing, including degassing and hot extrusion. Metallurgical examinations suggest

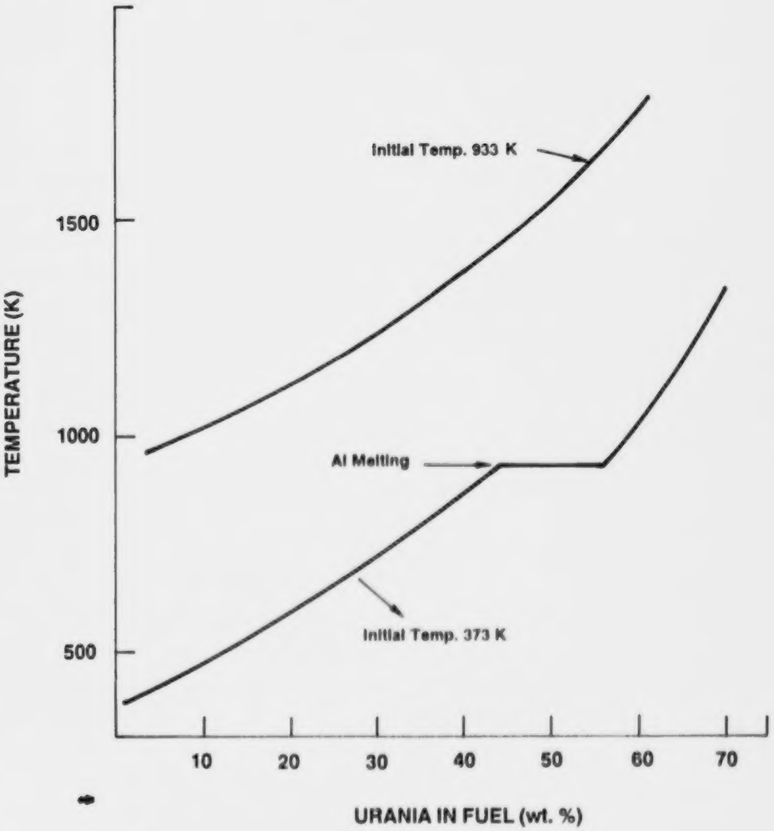

FIGURE F.1 Adiabatic temperature rises produced by the reaction of U_3O_8 with Al at initial temperatures of 933 and 373 K.

that some reduction of the U_3O_8 particles to U_4O_9 occurs at this time [23, 24]. The reaction of U_4O_9 with aluminum is significantly less exothermic and will produce less dramatic temperature excursions than does the reaction of U_3O_8 with aluminum. Furthermore, metallurgical examinations of irradiated cermet fuel suggest that during operation further reductions of the uranium oxide to a uranium-aluminum intermetallic compound and Al_2O_3 occurs [23]. The reduction processes during manufacture and operation reduce, of course, the potential for chemical heating posed by the cermet fuel. However, no safety analyses have been done to show how completely the U_3O_8 must be reduced before the chemical heating effects of the reaction are negligible. The committee believes there may be merit in assuring that the reduction of the U_3O_8 is complete as part of the fuel manufacturing process. There is, however, evidence that complete reaction of the U_3O_8 with aluminum may yield a product whose performance as a fuel is less satisfactory as a result of so-called "breakaway swelling," swelling that damages the fuel [27].

The reaction of aluminum with U_3O_8 has long been known [11-19]. Some confidence that this reaction may not be of concern has been gleaned from the relatively high temperature of ignition of the reaction—1150 to 1273 K, which is above the melting point of Al. For the analogous reaction of Al with Fe_3O_4, it is believed that the onset of reaction is inhibited by the presence of an adherent Al_2O_3 film at the interface between the metal and the reactive oxide [20]. Above about 1150 K, it is shown that Al_2O_3 films on Al surfaces begin to lose their effectiveness as a barrier to surface interactions [21]. The effectiveness of the Al_2O_3 film at preventing reaction is sensitive to impurities. Trace quantities of NaCl will reduce the ignition temperature for the thermite reaction from about 1160 K to the melting point of aluminum—933 K [20]. There is some suggestion that magnesium will do the same. Some tests conducted recently at Savannah River Laboratory [22] have shown low-temperature reactions of aluminum with the surface of pressed compacts of $Na_2U_2O_9$ and $Na_6U_9O_{24}$. Innoculation of U_3O_8/Al mixtures with NaCl did not reduce the ignition until quite high loadings of chloride were achieved. Low temperature ignition of the metallothermic reaction has not been detected in experiments to date with irradiated fuel specimens [23]. Some evidence suggests that metallothermic reaction accentuates the release of fission products from the fuel [26], though this enhanced release is no worse than what would be expected from fuels currently used in the Savannah River reactors.

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Appendix G
Hydrogen Generation During Accidents at N Reactor: Technical Discussion

Maintaining the integrity of the confinement system is crucial to the safety strategy of N Reactor. But the confinement building of N Reactor is quite vulnerable to overpressurization. Estimated design pressure limits for the confinement amount to only 5 psig [1]. Elaborate venting strategies have been devised so that the N-Reactor confinement would not be subjected to excessive loads during depressurization of the reactor in the early stages of an accident. Nevertheless, the N Reactor would be quite vulnerable to internal pressurization, such as pressurization caused by the combustion of hydrogen during an accident.

The materials used in N Reactor—metallic uranium fuel, zirconium clad, graphite moderator, and boron carbide scram system—are all quite reactive toward steam. The inventories of reactive materials in N Reactor vastly exceed equivalent inventories in commercial nuclear power plants [2]. The reactions of steam with these materials would be potent sources of hydrogen. Should even small percentages of these materials react with steam, sufficient hydrogen would be produced to threaten local hydrogen combustion if not combustion throughout the containment. Indeed, recent calculations with the HECTR code [4,5] show that localized combustion that threatens the integrity of the confinement could occur with just the amount of hydrogen thought generated during the "hypothetical accident" at N Reactor.

Sufficient attention has been called to the vulnerability of N Reactor to hydrogen combustion [2,3] that the contractor is now committed to installing a hydrogen mitigation system [6]. The design of such a system requires an understanding of the rate of hydrogen generation, when such hydrogen generation occurs, and the total amount of hydrogen likely to be produced. The discussion below is directed at the analyses of these features of an accident at N Reactor. In particular, it focuses on the following:

1. The use of supposedly bounding "design-basis" accidents to assess the threat of hydrogen combustion at N Reactor, and
2. The adequacy of the analyses of hydrogen generation for the design-basis accident.

DEFINITION OF THE BOUNDING ACCIDENT

As a basis for designing a hydrogen mitigation system for the N Reactor, a supposedly bounding reactor accident in which hydrogen is generated has been identified. Initially, this accident was found to be identical to the hypothetical accident described in the plant safety analysis [1] except that for the sake of being conservative a doubling of the predicted amount of hydrogen generation was allowed [5]. Recently, a revised analysis of the hydrogen design-basis accident has been developed [8]. This revision defines a so-called "end state" derived from the preliminary results of an incomplete Level 1 probabilistic risk assessment (PRA) study of the N Reactor [9]. This PRA purports to show that the most likely severe accident sequences will involve failure of the emergency cooling system (ECS) for at least part of the reactor core but continued performance of the graphite and shield cooling system (GSCS), even though the latter is not a "safety grade" system [7]. The GSCS is particularly important because the accident is ultimately terminated due to the heat removal capability of this system.

The use of design-basis accidents that incorporate peculiar, supposedly conservative physical processes has not served the reactor safety community studying commercial nuclear power plants very well. In fact, the practice is being abandoned in severe accident analysis in favor of procedures involving realistic modeling of a wide range of accidents with considerably different etiologies. One reason for this change is that one or more of these more realistic accidents is typically found to pose a more severe threat than the nonmechanistically described design-basis accident. In any case, there is no doubt that, taken in combination, the realistic accidents pose more formidable challenges to reactor safety systems than the design-basis accidents.

The design of the hydrogen mitigation system would benefit from the lessons learned by commercial reactor safety analysts concerning design-basis accidents. In that connection, the belief that the accident currently used for the design basis of the hydrogen mitigation system is bounding is not at all transparent. Accidents involving failure of the GSCS and degraded performance of the emergency cooling system have previously been described and would seem to involve (1) more extensive fuel melting, (2) higher temperatures, and (3) more prolonged duration than the current design-basis accident [10]. Though analyses of hydrogen generation for these accidents have not been provided, it seems likely that they would show far more extensive and perhaps more rapid hydrogen generation. Recent analyses of combustible gas generation during accidents involving failure of the GSCS seem based on undefended assumptions concerning the availability of steam and neglect of steam graphite reactions [47].

Confidence in the reliability of the GSCS based on preliminary results of the recent PRA study [9] may be misguided. There appear to be no specific reliability data for this unique system. The assumption in the PRA of failure rate probabilities on the order of 10^{-4} for this system would therefore appear to be based on generic data. Furthermore, the study does not consider external events, such as seismically induced accidents, to which the GSCS is vulnerable. The piping network of the GSCS is also threatened by the radiation-induced expansion of the graphite moderator stack. Indeed, four ruptures of the GSCS piping have been attributed to strains caused by offsets in the graphite blocks [7].

In summary, the committee believes that more extensive accidents than the current design-basis accident for N Reactor may be credible. These accidents would likely have substantially different hydrogen generation histories than the design-basis accident and would therefore place different requirements on the hydrogen mitigation system for N Reactor.

ANALYSES OF HYDROGEN GENERATION

To determine the rate and extent of hydrogen generation during accidents at N Reactor, the contractor has considered only the steam reactions with the zirconium clad and uranium fuel [8]. These reactions are assumed to take place in a surplus of steam and to follow parabolic reaction kinetics. These assumptions, together with the assumed successful performance of the GSCS, produce an unusual and highly specific hydrogen generation history, as shown in Figure G.1. The rate of hydrogen generation rises to a sharp maximum as an initial layer of product oxide is formed on reactive metal surfaces. Inhibition to the reaction, provided by the growing oxide layer, compensates for the effects of the slowly rising fuel temperature, and the rate of hydrogen production falls dramatically from its peak. As fuel reaches melting, a hypothesized reduction in the surface area of the melting fuel that is available for reaction with steam assures that there is no acceleration in the production of hydrogen. A later peak in the hydrogen generation is produced when regions of the reactor fuel subjected to lower heating rates go through similar cycles.

The hydrogen generation histories predicted for accidents at N Reactor are therefore highly dependent on the reaction kinetics of zirconium and uranium with steam. In the discussion that follows, the committee examines the analyses of these reaction kinetics and the potential for steam reactions with other materials in the N-Reactor core.

Zirconium-Steam Reactions

Steam reactions with the zirconium clad and process tubes during accidents at N Reactor are treated as obeying parabolic kinetics. The kinetic expression used for the analyses is one developed by Baker and Just [11]:

$$(\Delta W)^2 = 4.13 \times 10^6 \exp[-22900/T]\, t,$$

where ΔW = weight gain by oxidation (mgO/cm^2), t = time (s), and T = absolute temperature (K). This rate expression was devised for zirconium based on tests in the temperature range of 1270 to 2100 K. It is generally accepted that both Zircaloy-2 and Zircaloy-4 obey steam reaction kinetics similar to those followed by zirconium [12,13].

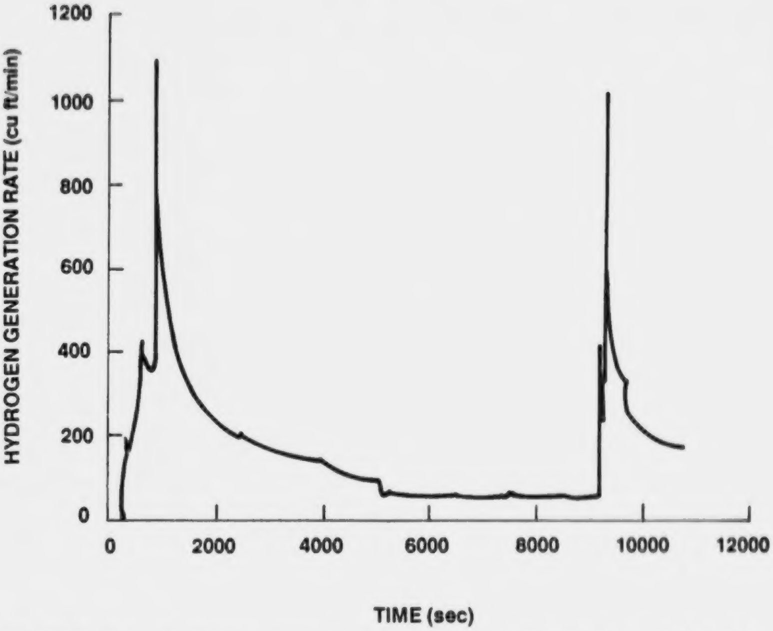

FIGURE G.1 Hydrogen generation rate as a function of time for the hydrogen mitigation system design basis.

The Baker-Just rate expression must be extrapolated to lower temperatures for N-Reactor accidents. It has been found [14] that at about 1163 K the kinetics of zirconium oxidation undergo a change caused by the monoclinic-to-tetragonal phase change of the substoichiometric zirconium oxide reaction product. Ianni [15] has found the following reaction rate expressions:

$$(\Delta W)^2 = 6.79 \times 10^3 \exp[-14595/T]\, t \text{ for } 298 < T < 1163, \text{ and}$$

$$(\Delta W)^2 = 4.40 \times 10^4 \exp[-16860/T]\, t \text{ for } 1163 < T < 1800.$$

Urbanic and Heidrick [13] have confirmed these rate expressions and have hypothesized another transition at 1813 K caused by a tetragonal-to-cubic phase change. Prater and Courtright [16] have developed a kinetic expression for the steam oxidation of molten zirconium:

$$(\Delta W)^2 = 3.1 \times 10^5 \exp[-16610/T]\, t,$$

where T > 2123 K.

Figure G.2 shows the parabolic rate constants for the various kinetic expressions as functions of temperature. Though Baker-Just kinetics are widely viewed as conservative, this is true only for the temperature regime pertinent to accidents in commercial nuclear power plants. Extrapolation of the Baker-Just rate expression to temperatures below about 1400 K leads to an underprediction of the rate of steam reaction with zirconium. It is, of course, in this low-temperature region where most of the steam-zirconium reaction would occur during accidents at N Reactor [8].

Quite clearly, analyses of hydrogen generation for the design of the hydrogen mitigation system need to be revised to take into account more accurate kinetics.

Although the precise values of the parabolic rate constants are important in the prediction of hydrogen generation from zirconium-steam reactions, of far greater significance is the sensitivity of the reaction kinetics to the microstructure and integrity of the oxide layer produced on the zirconium. The molar volume of zirconium oxide is, of course, much greater than that of zirconium (Bedford-Piling number in the range 1.56 to 1.41), so the oxide is internally strained when it forms on the metal. This strain is only partially reduced by the expansion of the metal as it dissolves oxygen. Excessive strains will eventually cause the oxide to rupture, and this will initiate a sudden acceleration in the rate of reaction as some of the inhibition provided by the oxide is lost.

Rupture of the oxide coating has not been shown to be an especially troublesome problem in studies of zirconium oxidation kinetics using flat specimens at isothermal conditions. But during N-Reactor accidents, nonisothermal oxidation would take place simultaneously on concave and convex surfaces. These surface geometries can exacerbate the innate strains of oxide growth [28]. On convex surfaces, poorly adherent oxide layers can buckle to relieve the compressive stresses (see Figure G.3). More adherent oxide layers, as growth of the layers continues, develop tensile stresses and fail by cracking. The effects are made worse by the monoclinic to tetragonal phase change. On concave surfaces, the initial compressive stresses grow with oxide growth, and rupture of the oxide can occur by shear (see Figure G.3). Contortion

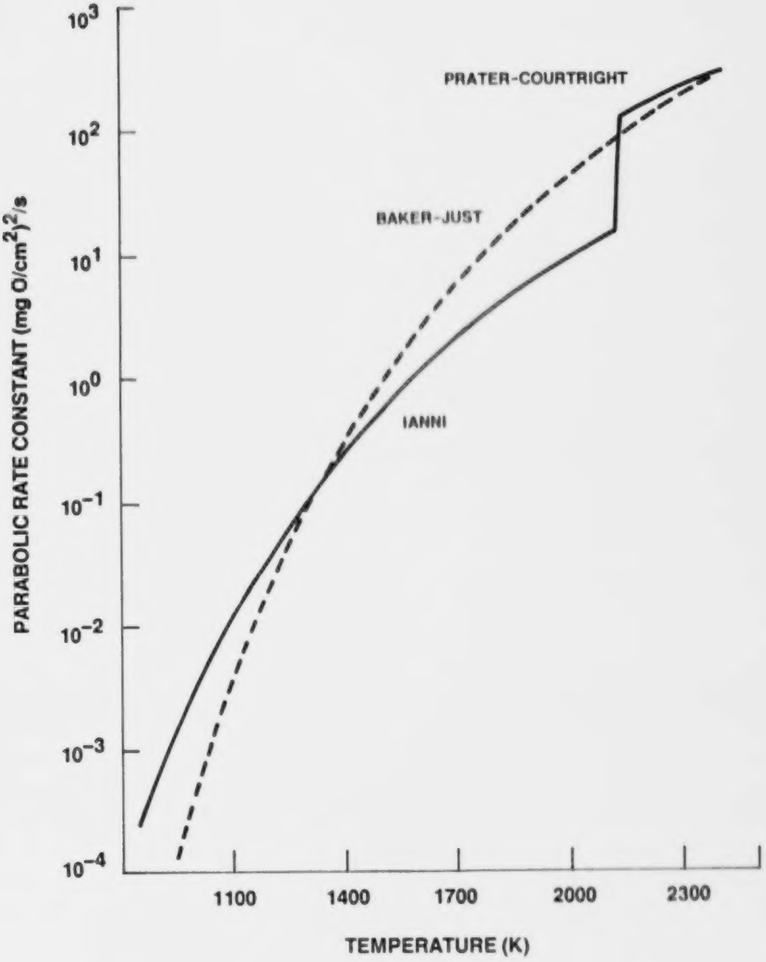

FIGURE G.2 Comparison of parabolic rate constants from various studies of zirconium oxidation in steam.

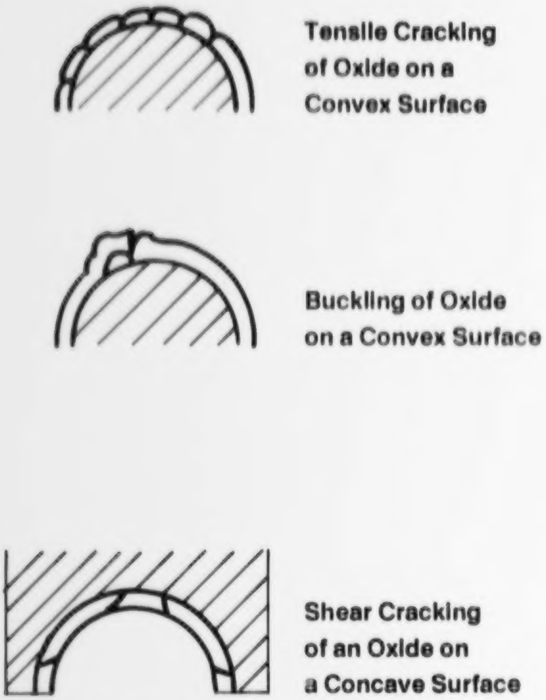
Tensile Cracking
of Oxide on a
Convex Surface
Buckling of Oxide
on a Convex Surface
Shear Cracking
of an Oxide on
a Concave Surface

FIGURE G.3 Modes of oxide layer rupture during oxidation of convex and concave zirconium surfaces.

and deformation of the clad and fuel observed in experiments [21] would make oxide layer rupture by these processes quite likely.[1]

Thermal transients are also known to rupture zirconium oxide layers on zirconium. For small temperature changes, fissures develop in regions of reduced oxide plasticity caused by anisotropy in the oxidation [29], and such inhomogeneity in the oxidation of N-Reactor fuel and process tubes would appear to be difficult to eliminate (see Appendix E).

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[1] There has been relatively little concern with phase transformations and oxide stresses in the analysis of steam-zirconium reactions during commercial reactor accidents. This is because the higher heatup rates of fuel in light-water reactor accidents lead to very little oxide being formed at low temperatures where the oxide is brittle. At the high temperatures of commercial reactor accidents, the zirconium oxide layers are sufficiently ductile to relieve strains that would cause oxide failure during N-Reactor accidents.

Large cooling transients could be induced in the N-Reactor fuel by "chugging" of coolant into tubes during partial performance of the emergency cooling system [10]. Such large changes in temperature can induce compressive stresses of a magnitude given by [17]:

$$\sigma = \frac{E_{ox}\Delta T(\alpha_{ox} - \alpha_M)}{1 + 2(E_{ox}/E_M)(h_{ox}/h_M)}$$

where

σ = stress in oxide;
E_{ox} = Young's Modulus of the Oxide;
E_M = Young's Modulus of the Metal;
ΔT = temperature change;
α_{ox} = thermal expansion coefficient of the oxide;
α_M = thermal expansion coefficient of the metal;
h_{ox} = oxide thickness; and
h_M = metal thickness.

Compressive stresses such as these will cause rupture of the oxide layer by spalling, which could be catastrophic.

The assumption of an uninterrupted progression of parabolic kinetics for steam reactions with zirconium during accidents at N Reactor needs to be defended. There are a variety of mechanisms that can cause disruption of the oxide, acceleration of the oxidation, and concomitant hydrogen production. Indeed, what data there are [46] show that hydrogen generation for N-Reactor fuel varies with time to the power 0.72 rather than the power 0.5. This is a substantial deviation from the parabolic kinetics assumed in the hydrogen design-basis accident analysis. Moreover, these limited tests did not involve all the factors that can disrupt reaction kinetics during N-Reactor accidents. The uncertainty in hydrogen production caused by these processes needs to be recognized in the analyses of N-Reactor accidents and in the design of a hydrogen mitigation system.

Kinetics of the Uranium-Steam Reaction

For the analysis of hydrogen generation during N-Reactor accidents [8], the uranium-steam reaction is assumed to follow the parabolic kinetics developed by Wilson et al. [18]:

$$(\Delta W)^2 = 9.94 \times 10^4 \exp[-9361/T]\, t \text{ for } 873 < T < 1473 \text{ K, and}$$

$$(\Delta W)^2 = 8.10 \times 10^5 \exp[-12600/T]\, t \text{ for } 1473 < T < 1873 \text{ K.}$$

These rate expressions were derived from isothermal studies. Exposure times were less than 200 minutes and usually less than 90 minutes.

Baker [19] also derived parabolic kinetics for the uranium-steam reaction from nonisothermal quenching experiments:

$$(\Delta W)^2 = 4.43 \times 10^6 \exp[-15000/T]\, t.$$

Initial temperatures of specimens in these experiments were 870 to 4000 K.

Other investigators do not agree that use of parabolic rate expression for the reaction of steam with uranium metal is appropriate. Both Hopkinson and Scott [20] found linear kinetics for temperatures up to 1150 K. At temperatures between 1150 K and the melting point of uranium, parabolic kinetics were obtained for reaction periods of up to 1 to 2 hours. Thereafter, linear behavior developed. Linear kinetics were observed for the reaction of molten uranium.

Again, it is noteworthy that what little data there are on steam oxidation of N-Reactor fuel do not confirm parabolic kinetics [46].

Obviously, there are substantial uncertainties in the kinetics of uranium oxidation. Replacing the parabolic rate expressions used in the N-Reactor accident analysis with linear kinetic expressions would have profound effects on the rate of hydrogen generation. Inhibition to the rate of reaction by the growth of the oxide product implied by parabolic kinetics would disappear as would the calculated fall in the rate of hydrogen generation. Prudence would dictate that allowance should be made for this uncertainty in the analysis of hydrogen generation for the design of a hydrogen mitigation system.

It is further assumed in the analysis that once the uranium fuel melts, there is a reduction in the surface area available for reaction to 10 percent of the value prior to melting [8]. This assumption is poorly supported by available data that suggest that the melting fuel exhibits a foaming behavior, which would enhance the surface area available for reaction [21]. This foaming and enhancement of surface area is probably caused by an acceleration in the release of volatile fission products as the fuel melts.

Large uncertainties must be ascribed to the hypothesized reduction in surface area available for oxidation, and therefore it is only prudent to examine less benign possibilities. Such an examination should include the possibility that molten uranium could locally attack corroded and flawed process tubes and drain into the graphite stack. This would not only enhance the surface area available for reaction with steam, but also produce additional heating from the carbon reaction with uranium.

Steam Oxidation of Graphite

The N-Reactor graphite moderator stack consists of about 1700 tons of graphite surrounding the fuel. Its purpose is to moderate neutrons. Use of graphite as a moderator, probably more than any other feature of the N Reactor, has attracted attention because of the superficial similarity that it represents to the Soviet RBMK reactor at Chernobyl, which also used graphite [22]. Though accidents of the precise type experienced at Chernobyl are not physically possible at N Reactor, the presence of such a large mass of combustible graphite cannot be ignored. The possibility that the inerting of the reactor vault might not be maintained in an accident and that the graphite might burn as air is drawn into the vault is discussed elsewhere in the report. The focus here is on reactions of steam with graphite to produce hydrogen and either flammable carbon monoxide or nonflammable carbon dioxide:

$$H_2O + C(\text{graphite}) \rightarrow CO + H_2$$

$$2\,H_2O + C(\text{graphite}) \rightarrow CO_2 + 2\,H_2$$

Though these reactions are endoergic, they are quite possible in a thermochemical sense and consequently could, in principle, contribute to the threat of combustion in the N-Reactor confinement.

The reactions of steam with graphite have been neglected in defining the design basis for the hydrogen mitigation system for N Reactor [8]. The rationale for having failed to treat these reactions is based on two points:

1. With the GSCS intact during the accident, temperatures within the graphite are kept low.
2. The rates of steam reactions with graphite are low.

The first of these considerations was discussed earlier. The kinetics of steam-graphite reactions are considered here.

The rates at which steam will react with graphite have often been studied. Typically, three regimes for the reaction are defined [22]. At elevated temperatures, the rate of reaction is controlled by mass transport effects. At low temperatures (< 1000 K) the rate of reaction is limited by the intimate surface reactions of oxidant (steam) with graphite. Analyses of N-Reactor accidents with the GSCS intact suggest that this low-temperature regime is the most pertinent, and chemical kinetics control the rate of reaction. Typically, the rate of steam-graphite reaction is formulated in terms of Langmuir-Hinshelwood kinetics based on the reaction steps:

$$H_2O(g) + S \Leftrightarrow H_2 + [O]_S$$

$$[O]_S \rightarrow CO + S,$$

where S designates an active surface site and $[O]_S$ designates an occupied surface site. The rate expression is [24]:

Rate [grams carbon gasified/gram carbon-second] =

$$\frac{K_1(T)\, P_{H_2O}\, F_{CAT}\, F_{BUR}}{1 + K_2(T)\, P_{H_2}^{N} + K_3(T)\, P_{H_2O}},$$

where P_x = partial pressure of x in the gas, and $K_i(T)$ = rate parameters for i = 1 to 3. The rate expression is empirically modified by the parameter N (typically N is about 0.75) to account for lost active sites. Geometrical evolution is accounted for by the parameter F_{BUR}. Catalysis of the reaction is described by the parameter F_{CAT}. More complete rate expressions that account for the effects of product gases have been developed by Ergun [25].

Analyses of the reaction presented in the NUSAR document [1] were based on kinetic parameters recommended by Woodley [26]. These parameters are compared in Table G.1 to parameters used in a more recent study for N Reactor [27] and from other studies in the literature [24]. The scatter in the data, none of which are for the exact graphite used in N Reactor, is likely to

TABLE G.1 Kinetic Parameters for the Steam Oxidation of Graphite

Parameter	Value from Reference [26]	[27]	[24]
$K_1(T)$	$5.94 \exp[-16460/T]$	$2.97 \times 10^4 \exp[-20584/T]$	$9.12 \times 10^7 \exp(-32960/T)$
$K_2(T)$	$9.4 \times 10^{-11} \exp[30599/T]$	$0.0166 \exp[14394/T]$ [a]	$4.49 \times 10^{-4} \exp(14398/T)$
$K_3(T)$	$7.06 \times 10^{-16} \exp[39909/T]$	$0.0531 \exp[13840/T]$	$1.32 \times 10^{-4} \exp(15805/T)$
N	1	0.75	0.75

[a] Set to zero for accident analysis.

by irradiation by gamma rays during reaction, both of which are reported to enhance the rate of reaction [31].

A recent analysis of the steam oxidation of N-Reactor graphite when the GSCS is not functional [27] indicates that about 5.7 percent of steam flowing through the graphite block will react with the graphite. If the steam flow is that which can be generated by the helium purge system [30], then hydrogen would be produced at the rate of about 60 kg/hr (1980 cu.ft./min) and carbon monoxide would be produced at a rate of about 840 kg/hr [27]. These rates are similar to the rates of hydrogen production for just the steam-metal reactions considered in N-Reactor accident analyses to date. This result stands in sharp contrast to a recent analysis of the consequences of failure of the GSCS prepared by the N-Reactor staff [47]. In their analysis no combustible gas generation caused by steam reaction with graphite was included despite the loss of the cooling provided by the GSCS. It is difficult to reconcile this result with available information on steam-graphite reactions or the heat up of the N-Reactor graphite when cooling is lost.

Reactions of steam with graphite, when the reaction is controlled by the rate of chemical processes rather than mass transport, can be accelerated by catalysis [32,22]. Such catalytic activity has not been considered in the analysis of accidents at N Reactor, even though rather mundane materials can act as catalysts. For instance, iron, cobalt, and nickel have been found to accelerate the steam reactions with graphite by factors of 32, 27, and 19, respectively [34]. In view of the age and the nature of operation of the N Reactor, it is difficult, if not impossible, to preclude the presence of such catalysts.

Perhaps even more significantly, fission products are known to catalyze the steam reaction. Alkaline earth oxides, for example, are known catalysts [36-38]. Everett et al. [37] found that at 1120 K barium and strontium salts at concentrations of 0.05 percent in the graphite accentuated the rate of steam reaction with graphite by factors of 1000 and 130, respectively. It appears that barium and strontium lower the activation energy for reaction [38]. There is evidence that Ru, Rh, and Pd will also act as catalysts [39,40]. Lithium, sodium, and potassium catalyze the steam reaction with carbon by a reaction process presumed to be [35]:

$$M_2CO_3\ (l) + 2\ C \rightarrow 2\ M\ (g) + 3\ CO$$

$$2\ M\ (g) + 2\ H_2O \rightarrow 2\ MOH\ (l) + H_2$$

$$2\ MOH\ (l) + CO \rightarrow M_2CO_3\ (l) + H_2$$

where M = Li, Na, K. In view of the similarity of potassium and cesium chemistries, it is likely that cesium, too, will catalyze the reaction.

It appears, then, that catalysis of the steam reaction with graphite, if not initiated by impurities already in the N-Reactor graphite, could be initiated once core degradation leads to fission product release and fission products contact the graphite. The catalysis of graphite reactions could lead to production of both CO and H_2 at rates comparable to those currently considered in the design basis accident. Such combustible gas generation will certainly occur if the GSCS has failed and may be significant even if it has not failed.

In view of the above, neglect of the graphite-steam reactions in N-Reactor accident analyses is questionable.

The Oxidation of Boron Carbide

The N Reactor uses boron carbide both in control rods and in the backup scram system. Boron carbide will react with steam to form hydrogen [41,42]:

$$B_4C + 6\ H_2O \rightarrow 2\ B_2O_3 + C + 6\ H_2.$$

To date, this reaction has not been considered as a potential source of hydrogen in analyses of N-Reactor accidents.

The N Reactor's boron carbide control rods are well-sheathed. Consequently, this boron carbide would experience little exposure to steam, except as a consequence of a very catastrophic event that disrupted the core.

Boron carbide used for the backup scram system is in the form of unsheathed spheres (with diameters of about 0.95 cm) that are dropped into 107 vertical channels in the N-Reactor core [1]. There are liners on the channels, although they do not hermetically isolate the channels.[2] It must be presumed that under accident conditions these channels are possible vent pathways for steam from the core, and that the boron carbide balls would be bathed in steam. The reaction of boron carbide with steam is readily detectable at temperatures as low as 250°C [39]. Definitive analyses of temperatures that can develop within the ball channels under accident conditions are not available. It appears that low temperatures can be maintained in the core only if the GSCS remains functional during the accident. As noted earlier, this system is not of safety grade.

The reaction of boron carbide with steam is kinetically slowed by the formation of a protective outer layer of the condensed reaction product, B_2O_3. The reaction would be expected to obey parabolic kinetics initially. At somewhat elevated temperatures, molten B_2O_3 (which has a melting point of 733 K) would drain from surfaces, thus limiting the protection provided by the product layer. Reaction of B_2O_3 to form vaporous boric acid:

$$B_2O_3 + H_2O \rightarrow 2\,HBO_2\ (g)$$

$$B_2O_3 + 3\,H_2O \rightarrow 2\,H_3BO_3\ (g)$$

would further limit the growth of the product layer.[3] Overall, paralinear kinetics would be expected for the reaction of boron carbide with steam. Relatively little quantitative data are available on the reaction kinetics. There is evidence of some tendency for localized or "pitting" corrosion of boron carbide by steam [41]. An empirical study of the corrosion, using B_4C balls of the type employed at N Reactor, has been conducted and shows the reaction to proceed rapidly above about 1000 K [45].

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[2] Prior to installation of the liners, some boron carbide balls were "lost" within the N-Reactor graphite stack. It is presumed here that these "lost" balls constitute a trivial mass of reactive material.

[3] The neutronic consequences of boron relocation by this reactive vaporization process are not examined here.

229

Litz and Mercuri [43] found the rate of steam reaction with boron carbide to be:

$$\text{Rate}\ \left(\frac{\text{grams oxidized}}{\text{gram } B_4C - \text{hr}}\right) = 47\ P_{H_2O}\ \exp\left[-5360/T\right]$$

where P_{H_2O} is the steam partial pressure in atmospheres. Based on this rate expression about 30 Kg of hydrogen would be produced per hour if the boron carbide balls in N Reactor were flooded with steam at 800 K. This hydrogen would supplement that produced by steam reactions with the fuel, clad, and graphite.

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COMMISSION ON PHYSICAL SCIENCES, MATHEMATICS, AND RESOURCES
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COMMITTEE TO ASSESS SAFETY AND TECHNICAL ISSUES AT DOE REACTORS

August 26, 1987

Secretary John S. Herrington
Department of Energy
Washington, D.C. 20585

Dear Secretary Herrington:

On May 22, 1987, Dr. James F. Decker, the Acting Director of the Office of Energy Research, asked the Committee to Assess Safety and Technical Issues at DOE Reactors to review the technical adequacy of the Department's proposed approach to hydrogen mitigation at the N Reactor. I am writing on behalf of the Committee to respond to this request. The Committee will shortly submit another report that will discuss a range of other issues associated with the Department's production reactors.

The scope of the Committee's review of the proposed approach to hydrogen mitigation has been limited. As requested by the Department, we have focused on the overall advantages and disadvantages of the proposed approach. We have not performed a detailed design review. We have not reviewed operating procedures or training requirements, which are critical to the successful implementation of the approach. Moreover, we necessarily have relied on the accuracy of the detailed calculations in the materials that were submitted to us for review. Finally, our review has not addressed some of the more general assertions included in the contractor's documents. A list of the materials provided to the committee is appended.

In Part I of this letter we focus on the nature of the problem, the Department's proposed approach, and our principal conclusions. In Part II, we turn to a detailed discussion of specific issues. Both parts of the letter presuppose a basic familiarity with the relevant characteristics of N Reactor. These are described more fully in the materials provided to us for review.

I.

The protection of the public in the event of a severe accident at the N Reactor depends on maintaining the integrity of the reactor's confinement system -- the safety system that is designed to attenuate the release of fission products in the event of an accident. Because the confinement has not been designed for -- and is unlikely to be able to withstand -- hydrogen burns of any significant size, it is important that combustible concentrations of hydrogen be prevented within the confinement, except in very localized areas near the hydrogen source. In order to evaluate this issue, the contractor for the facility has conducted a series of analyses in which certain defined releases of hydrogen were assumed to occur in a number of locations within the confinement. The results of these analyses indicated to the contractor that, for the assumed rate and quantity of hydrogen released, there are only two locations in the confinement where combustible limits could be achieved -- the pipe-barrier space and the pressurizer-penthouse area. In other areas of the building, the contractor predicted that mixing would be sufficient to prevent a flammable concentration from being reached. The contractor further concluded that even if a burn were initiated in the pipe-barrier space, the increase in pressure would be dissipated in the larger volume of the confinement building and the confinement function would not be defeated. In order to mitigate the potential problem in the penthouse and to provide additional protection from the release of hydrogen in a severe accident, the contractor has developed a concept for a hydrogen mitigation system.

The proposed concept is intended to satisfy four basic functional requirements:

- Provide adequate mixing between subcompartments of the building to assure that burnable concentrations of hydrogen do not accumulate.
- Monitor hydrogen concentrations at various locations within the building.
- Fill the confinement with inert gas after the initial release of steam to assure that burnable compositions are precluded in the long term.
- Provide an exhaust system to displace air to support the inerting function and to establish a lower pressure in the building than outside the building, thus preventing outleakage through the walls and bypass of the filter system.

The first function is to prevent burnable concentrations of hydrogen. The second function provides information to plant staff that could be useful for identifying and mitigating severe-accident situations. The latter two functional requirements could be just as important in that, if properly implemented, they could provide a margin of protection in the event that releases of hydrogen are greater than the assumed hydrogen source.

The contractor considered alternative approaches to hydrogen mitigation. Preinerting and distributed ignition systems, which are methods for hydrogen control used in commercial reactors, were found not to be appropriate for the N-Reactor confinement system. The initial venting of the building atmosphere that occurs in the proper operation of the confinement system would defeat any preinerting system, and the confinement might not be able to withstand pressure pulses that could occur with a distributed ignition system, particularly for accidents involving large releases of hydrogen. This latter conclusion is consistent with the observation of pressure rises of many pounds per square inch in tests of ignition systems performed by the Electric Power Research Institute in a large-scale facility at the Nevada Test Site. Within the bounds of reasonable cost, there is no apparent preferred alternative to the concept proposed by the contractor.

The Committee's overall assessment of the general approach to hydrogen mitigation is favorable. In indicating its agreement with the general approach to hydrogen mitigation at N Reactor, the committee is not rendering a judgment as to the adequacy of the proposed system. Any judgment as to the adequacy of the system must be guided by an assessment of the general approach, careful analyses of the system under various hypothetical accident scenarios, evaluation of a detailed design, and construction and operation in compliance with appropriate safety standards. Because the analysis is only at a preliminary stage, this letter addresses only the first of these necessary factors.

The basic strategy of forced mixing to remove the potential for hydrogen pockets, monitoring of hydrogen at key points within the confinement, and the activation of an inerting/exhaust function appears to us to be sound. Indeed, the approach may provide a margin to accommodate some uncertainty relating to the amount of hydrogen generated in an accident. However, as discussed further in this letter, there are aspects of the underlying analyses and certain potential limitations of the proposed system that deserve further careful evaluation.

The principal aspects of the approach that merit further study or improvement may be briefly summarized:

- The system design is premised on a specific accident scenario. In the Committee's view, the predicted performance of the mixing and inerting systems should be examined for a broader spectrum of accident scenarios and release rates for hydrogen. For example, the assumption of the continuing operability of the Graphite and Shield Cooling System (GSCS) in the event of an accident needs a thorough examination, particularly in terms of any possible degradation of the GSCS in an accident.

- In the course of examining a broader spectrum of accident scenarios and hydrogen release rates, the contractor should extend the mixing calculations to address localized mixing and combustion. In addition, the contractor should examine the geometric configurations near possible release points in order to assure that localized concentrations of hydrogen do not permit local detonations that could challenge the confinement or other safety systems.

- The capability of the filter system to withstand the loading of aerosols in an accident should be reviewed and, if necessary, the capability of the system should be upgraded.

- The survivability of vital sensors and mitigation equipment should be assessed. The equipment must be capable of operating in severe-accident environments characterized by wide variations in thermal-hydraulic, radioactive and inert aerosol loads, and high radiation fields.

- The operation of the system necessitates the discharge of noble gases to the environment in the event of an accident. Although the operation of the system would decrease the risk of failure of the confinement following an accident, its operation would result in potential increases in whole-body doses to persons outside the plant. Means should be investigated to assure satisfaction of the dose limits of 10 CFR Part 100 without relaxing the speed with which inerting takes place. Such an investigation should take into account recent research that shows that the actual quantity and chemical form of the fission products released in a severe accident would be quite different from the source terms commonly assumed in such analyses in the past.

The Committee recommends that these concerns be resolved before the implementation of the proposed approach. The detailed design should be subjected to an independent review before hardware installation. Careful review of other aspects of the system, including operating procedures and training, is also warranted. In addition, prior to installation, the contractor should assess the effect of the proposed system on plant risk to determine whether the system would in fact result in lower risk to the public.

II.

There are a number of aspects of the proposed approach that warrant specific comment.

The Design Basis For The System

The accident scenario used as the design basis for the hydrogen-mitigation system was justified through a preliminary probabilistic risk assessment conducted by Los Alamos National Laboratory. In this scenario, the emergency core cooling system is assumed to fail following a large pipe break accident. An effort was made to be conservative in analyzing this accident and thus to provide for uncertainties, which at the present state of the analysis are substantial. The principal conservatisms in the analysis are the assumption of an unlimited supply of steam for the metal-water reaction (the source of the hydrogen) and the doubling of the hydrogen release that is calculated.

In our view the accident scenario that provides the design basis for the mitigation concept is not unequivocally conservative. We thus recommend that further consideration be given to alternative accident scenarios. Two assumptions in the current analysis clearly warrant increased attention:

- The analysis assumes that the cooling system for the graphite moderator (the GSCS) would be fully operational following an accident. The assumed continued operation of the GSCS reduces the calculated rate of hydrogen generation. This assumption should be validated from either a probabilistic standpoint (i.e., low probability of GSCS failure) or a mechanistic analysis (i.e., that the hydrogen source resulting from failure or degradation of the GSCS can be accommodated by the mitigation system).

- The analysis assumes that only a small fraction of the surface area of the uranium fuel is available for oxidation after melting. This serves to limit the extent of the metal-water reaction and thus the amount of hydrogen that is evolved. This assumption could be a major source of non-conservatism in the present analysis. More model development and experimental validation are warranted. Similarly, further consideration should be given to the possibility of catalytic enhancement of the reaction of graphite and steam, to the potential for fracturing of the zirconium oxide layer that is formed on the fuel during the presumed accident, and to the potential underestimation of the kinetics of zircaloy oxidation.

Although the Committee has concerns that the design basis for the hydrogen mitigation approach may not be demonstrably conservative, we believe that the approach provides a margin which, if maintained in implementation, may accommodate higher hydrogen release rates.

Hydrogen Monitoring System

The monitoring of hydrogen concentration within subcompartments of the confinement building is important because it provides information on localized conditions that can guide decisions in an emergency. The hydrogen monitoring system that is currently being installed measures the concentrations of hydrogen and oxygen electrochemically in ten sampling locations in key regions of the confinement volume. The system will require at least 30 minutes to cycle through the ten sample points to sample the entire confinement atmosphere.

The detection of hydrogen should not govern the activation of the hydrogen-mitigation system; the completion of the inerting function requires a number of hours and if it were not started until after hydrogen production is observed, it might be activated too late for the system to be effective. Furthermore, the 30-minute cycling interval is definitely longer than desired even if the measurements are not the sole guide for the activation of the hydrogen-mitigation system. Given the importance of timely information on the status of the system in an emergency, the capability for more rapid measurement should be sought and achieved. In addition, it is important that a monitor be placed near the point of release of the vent lines that connect with the dump tank for the reactor's depressurization valves, since this is the most likely point of release of hydrogen from the primary system for certain kinds of accidents.

Hydrogen Mixing System

Studies by the contractor have indicated that hydrogen concentrations after an accident would be expected to exceed the combustible limit (4 percent) in the small pressurizer penthouse if there were no forced mixing. The focus of the design for the mixing system, therefore, is to prevent the buildup of hydrogen in the penthouse. Redundant mixing fans with a capacity of 25,000 to 40,000 cubic feet per minute would remove air from the front face of the reactor and discharge it to the penthouse. Return flow through cross vents would provide for mixing of the major confinement volumes.

If the assumed hydrogen source term and the mixing calculations are reasonably accurate, the proposed mixing capacity should be fully adequate to prevent burnable concentrations (other than very localized concentrations at the break location) within the confinement. However, it is unclear how much margin exists in other regions, such as the pipe gallery and steam generator cells, in the event that hydrogen production rates should exceed the postulated hydrogen release. Hence, the Committee recommends more thorough evaluation of alternative accident scenarios and the uncertainties associated with calculations for hydrogen pocketing and stratification.

The basis for the concept of the mixing system derives from analytical simulations of hydrogen mixing in the N-Reactor confinement which indicate to the contractor that, with the exception of the pipe-barrier space and penthouse areas, natural-convection currents would be sufficiently strong to yield an essentially uniform concentration of hydrogen throughout the confinement volume. Validation of these analytical tools and their applicability to the problem of hydrogen mitigation at N Reactor (particularly with regard to the different scales of hydrogen stratification that may occur in the confinement) remain to be demonstrated. It is well known that calculated results for mixing processes in large open compartments are influenced by the degree of nodalization for lumped-parameter models and are subject to numerical diffusion errors for finite-difference codes. The Committee also recommends that the geometric configurations near possible hydrogen release points from the reactor coolant system be reviewed to assure that concentrations of hydrogen do not permit local detonations that could threaten the confinement or other safety systems.

Inerting/Exhaust System

The proposed inerting system would consist of redundant trains of nitrogen, supplied from stored quantities of liquified nitrogen, at a rate of up to 20,000 cubic feet per minute. Activation would follow primary system depressurization and the associated steam venting from the confinement following an accident. In order to allow the inflow of nitrogen without overpressurization of the confinement, an equal flowrate of air must be exhausted. In the proposed system, which has a maximum capacity of 23,000 cubic feet per minute, a negative pressure differential is expected to be achieved (relative to the atmosphere) shortly following the initial venting of the confinement. This has the benefit of reducing the subsequent direct outleakage of radioactive material through the walls of the confinement. This type of outleakage is a major contributor to the calculated doses, particularly thyroid doses, that are estimated in the current N-Reactor safety analysis report.

The contractor's calculations at N Reactor indicate that at the maximum rate of nitrogen introduction, with accompanying displacement of air from the confinement, inerting could be complete (less than 5 percent concentration of oxygen) in about 3.5 hours. Some hydrogen would also be displaced from the confinement by the nitrogen, further delaying the time required to achieve burnable quantities.

The Committee has examined the inerting capability using a simple single-volume model and assuming instantaneous mixing. We applied the model to study a range of hydrogen release rates into the confinement. The calculation reveals that a margin is available to cope with larger releases of hydrogen than are generated in the design basis scenario. This margin, if it in fact were achieved in implementation of the approach, serves to alleviate somewhat the Committee's concern that the postulated accident that serves as the design basis is not unequivocally conservative.

In order to achieve the extra margin, however, the system must be well mixed and the exhaust and inerting rate must be high (20,000 cubic feet per minute). The contractor did not provide adequate information to enable us to assess whether the mixing condition would be achieved, particularly for the steam generator cells. And care must be taken in implementation of the system to assure that rapid inerting can be achieved without accompanying releases that might compromise public health.

The operation of the inerting/exhaust system following an accident would discharge noble gases to the environment more rapidly than would the existing design, resulting in potentially higher whole body doses to individuals outside the plant. There is a direct tradeoff between the benefit of preventing hydrogen explosions through inerting and the cost of increasing the doses from noble gases. The implementation of the approach must provide a rapid inerting capability and assure compliance with the dose limits of 10 CFR Part 100. In this connection, it is important to recognize that assuring compliance with 10 CFR Part 100 involves uncertainties associated with operation of the confinement system that were not analyzed in the material provided to the Committee (for example, the confinement vent valves, the spray system, the filters).

Implemention of the contractor's proposal will also require careful reevaluation of the filter system. Higher exhaust flow could increase the aerosol loading on the filters and, hence, increase the likelihood of filter failure. Moreover, the decreased residence time of aerosols within the confinement system after an accident would decrease the effectiveness of the confinement sprays in removing aerosols from the atmosphere and thereby increase further the threat to the filter system. The potential aerosol source term, the spray effectiveness in the removal of aerosols, and the failure criteria for overloading of the filters require careful examination by the contractor.

Other potential failure modes of the confinement system that stem from the activation of the hydrogen-mitigation system also warrant scrutiny. For example, the injection of cold nitrogen into the confinement in conjunction with operation of the fan exhaust system could reduce the pressure in the confinement, possibly challenging the structure. Careful examination of the interaction of the hydrogen-mitigation system with the confinement design and operation is necessary.

Finally, it should be recognized that, as with any inerting system, there is a possibility of accidental actuation. Careful controls must be employed to prevent inadvertent actuation of the system and to alert personnel located in the building in the event it occurs.

Design, Control System, Procedures, and Training

The hydrogen mitigation concept has not developed to the point at which the details of the design, the control system, the operating procedures, and the training programs can be assessed. It is clear, however, that successful implementation of the concept must rely on the careful development and implementation of these elements. Accordingly, we recommend a detailed independent review of each of these elements.

* * *

We hope that these comments are helpful. Please feel free to contact the Committee if we can provide any further assistance.

Sincerely,

Richard A. Meserve
Chairman

cc: Joseph F. Salgado
James F. Decker

Documentation Provided for Committee Review

1. UNI-4398, G. R. Franz, "An Overview of Hydrogen Control in N Reactor," May, 1987.

2. UNI-4397, F. J. Heard, et al., "N Reactor Safety Enhancement Interim Report - Hydrogen Generation and Thermal Analysis for the Hydrogen Design Basis Accident," April, 1987.

3. UNI-4381, G. S. Petersen, "Engineering Study - Hydrogen Mitigation Alternatives," April, 1987.

4. UNI-4332, L. D. Muhlestein, (HEDL) et al., "Hydrogen Mixing and Distribution in N Reactor Confinement - Interim Status Report," March, 1987.

5. UNI-4404, L. J. Kripps (EI International), "Regulatory Perspective on the Hydrogen Issue at the N Reactor," April 15, 1987.

6. UNI-4259, D. H. Shuford, "Functional Design Criteria 100-N Hydrogen Monitoring System," Project H-798, March 5, 1987.

7. UNI-4283, D. H. Shuford, "Conceptual Design Report 100-N Hydrogen Monitoring System," Project H-798, March 19, 1987.

8. UNI-4342, D. H. Shuford, "Conceptual Design Report Confinement Hydrogen Mitigation System," Project H-791, Draft.

9. UNI-4308, W. F. Zuroff, "Functional Design Criteria Confinement Hydrogen Mitigation System," Project H-791, Draft.

10. Memo, G. R. Franz to Distribution, "Hydrogen Mitigation Design Basis Source Term," dated April 2, 1987.

11. UNI-M-126, Project Management Manual, Section 12.0, Procedure P-14, "Design and Constructability Reviews."

12. HEDL-M-200, Engineering Manual, E.P. 4. 1, Design Verification.

13. UNI-I-111, D. W. Stack (LANL), "Preliminary Results of a Probabilistic Risk Assessment for the N Reactor,", April 10, 1987.

14. UNI-4400, G. R. Franz, "Hydrogen Monitoring and Control Recommendations," April 15, 1987.

15. UNI-3691, A. W. Barsell, et al., "Analysis of the Reaction of Steam with N Reactor Moderator Graphite," September 1986 (GA Technologies Inc., Report GA-C18610). NAS was provided a copy of this document January 1, 1987. We are providing a copy for your review.

16. UNI-4417, D. M. Ogden, et al., "Evaluation of the Proposed Hydrogen Mitigation System for N Reactor," to be published June 1987.

17. UNI-M-90, "N Reactor Updated Safety Analysis Report - NUSAR," 1978 as amended. This document is available at DOE-HQ and was previously sent to the NAS.

Appendix H
Structure of the DOE Safety System: Technical Discussion

This appendix describes the general system that is used to establish and assure the safety of the nation's defense production reactors.

The U.S. Department of Energy (DOE) has established a safety system for the production reactors that has three major elements:

- A safety objective;
- Orders that prescribe the means for achieving the objective; and
- A process for ensuring and verifying compliance.

The committee concludes, based on the examination that follows, that DOE's conception and implementation of each of these elements is less comprehensive, understandable, and consistent than is desirable.

The safety system is highly complex. No single document describes the structure of the system as a whole. Nor is there a single source for DOE's safety objective and operating standards. These derive not only from the Department's Orders, but also from legislation, field office directives, contracts, award fee evaluation plans, contractor performance goals, safety and hazards analyses, technical specifications, technical and process standards, test authorizations, and various types of appraisal reports. To attempt to capture the essence of the safety philosophy reflected in this large volume of material is not a trivial undertaking. The discussion that follows, therefore, is not intended to be comprehensive; it merely aims to illustrate some of the major features of the DOE safety system in order to provide a basis for some general conclusions.

SOURCES OF DOE'S SAFETY OBJECTIVE AND STANDARDS

DOE's safety objective and standards encompass a diverse set of requirements. They range from highly abstract, basic safety principles to narrow technical operating limits governing the day-to-day operation of the reactors. However, as discussed below, the overall system is built upon a single, fundamental premise that was established in legislation over 40 years ago.

Legislation

In operating the production reactors, DOE and its contractors must comply with a variety of federal statutes, including the Occupational Safety and Health Act, the National Environmental Policy Act, and the Clean Air and Clean Water acts, to name a few. However, DOE's safety system derives from the Atomic Energy Act of 1954, as amended [1].

The Atomic Energy Act of 1954, as amended, is the primary statute governing the ownership and control of nuclear materials and nuclear reactors, both commercial and defense-related. Yet most of the specific provisions of the Act—and all of the licensing and related regulatory requirements—apply solely to commercial reactors regulated by the Nuclear Regulatory Commission (NRC); they are not applicable to those operated by DOE. Moreover, other than charging DOE to assure the protection of public health and safety, Congress has left to DOE's discretion the manner in which this responsibility is to be achieved. Consequently, the system for ensuring the safety of the defense production reactors has developed quite separately, and differently, from that established for commercial reactors.

The nation's basic policy concerning the need for the production reactors is spelled out in a series of congressional findings in the Atomic Energy Act. The relevant provisions of the act state the following:

- The development, utilization, and control of atomic energy for military . . . purposes are vital to the common defense and security. [Sec. 2a]
- The processing and utilization of source, byproduct, and special nuclear material must be regulated in the national interest and in order to provide for the common defense and security and to protect the health and safety of the public. [Sec. 2d]
- Source and special nuclear material, production facilities, and utilization facilities are affected with the national interest, and regulation by the United States . . . is necessary in the national interest to assure the common defense and security and to protect the health and safety of the public. [Sec. 2e]

The objective of protecting the public health and safety appears in all subsequent legislation pertaining to DOE's mission. It was contained in the statute that established DOE's predecessor, the Energy Research and Development Administration (ERDA), wherein ERDA was charged with operating the agency ". . . to advance the goals of restoring, protecting and enhancing environmental quality, and to assure public health and safety." [2] And the same broad goal appeared in the DOE Organization Act of 1977, which stated that DOE's mission was to include ". . . incorporation of national environmental protection goals . . . and . . . the goals of . . . assuring public health and safety." [3]

It is clear, therefore, that DOE's overriding responsibility with regard to the production reactors is protection of public health and safety. It is important to recognize, however, that Congress gave DOE nearly complete discretion to determine how it should go about protecting the public.

Department Orders and Directives

The DOE regulates the production reactors by prescribing "Orders" that, as a result of contractual provisions, must be followed by the contractors. (Table 1.1 in Chapter 1 presents a list of the main safety-related DOE Orders currently applicable to the production reactors.)

The DOE Orders set forth requirements for the production reactors either directly, in the form of specific provisions in the Orders, or indirectly, by reference to standards established elsewhere. These requirements define how the Department expects public health and safety to be assured. They establish a set of prescribed activities and goals, including the identification of potential hazards and their consequences [4]; assurance that reasonable measures are taken to eliminate, control, and mitigate hazards [4]; reduction of identified safety and health risks, where possible, whether mandated by specific requirements or not [5]; and operation, maintenance, and modification of the reactors in accordance with standards consistent with those applied to comparable licensed reactors. [6] These various activities can be summarized in terms of two major safety objectives: (1) to establish comparability of the production reactors with commercial licensed reactors, and (2) to effectively identify, prevent, and mitigate production reactor risks. How these objectives relate to one another is unclear. What is clear is that the Department's approach depends entirely on judgments and interpretations of what such terms as "adequate protection," "reasonable measures," and "comparability" with licensed reactors may mean.[1]

The DOE Orders identify standards—programs, practices, procedures, and other guidelines—that the contractors are expected to follow to achieve the Department's objectives. These are of two principal types: "mandatory" and "reference" standards. DOE has standards covering the following general areas of reactor safety: training and qualification of reactor personnel, plant operations, safety system modifications, safety analysis, technical specifications, quality assurance, materials storage and handling, reporting requirements, unusual occurrences, emergency planning, administrative controls (internal review and appraisal system), fire protection, radioactive waste management, and radiation protection.

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[1] The Department's Orders and field organisation directives differ completely from the requirements imposed by the Nuclear Regulatory Commission on commercial reactors. The NRC's policy statements, rules, standard review plan, regulatory guides, criteria, and bulletins and orders are far more self-explanatory, detailed, comprehensive, and consistent than the requirements in the DOE Orders. Moreover, the NRC has a more elaborate and formal process for backfitting newly developed regulations on existing reactors.

The DOE Orders make limited reference to commercial reactor standards. Moreover, with the exception of those Orders governing reactor modification, in which the contractors are required to meet the NRC's standards for acceptable doses at the site boundary during normal operations and during accidents (10 CFR 100), the direction to meet NRC and commercial industry standards is generally either so qualified or so ambiguous as to raise doubt about the extent to which they are applicable. For instance, the training and qualification of reactor personnel are supposed to meet ANS standard 3.1 ". . . to the extent appropriate." [6] The relevant NRC regulatory guide ". . . shall also be considered." Plant modifications only have to meet the NRC's general design criteria (10 CFR 50) if and when DOE ". . . determines that safety can be significantly improved." [7] Contractors are required to document technical specifications, but the technical specifications themselves only have to be ". . . similar to those required for comparable facilities licensed by the NRC. . . ." [6] Although a DOE Order states that an ANSI/ASME industry standard for quality assurance programs ". . . is the preferred standard for quality assurance," the order elsewhere states that DOE encourages ". . . the judicious and selective application of elements of appropriate, recognized standards" for quality assurance programs. [8]

DOE Order 5480.4 specifically addresses the question of standards. It defines five sets of industry codes as "mandatory Environment, Safety and Health (ES&H) standards" that are required as a matter of DOE policy, but only if and when DOE determines that their application would increase safety. To our knowledge, no formal documentation of such a finding has ever been made. In addition, DOE Order 5480.4 lists many "reference ES&H standards," which are described as "references on good practice." But the reference standards have no specific or prescribed relationship to the DOE regulatory process nor are they implemented in a timely fashion.

Contracts

Du Pont operates the Savannah River reactors under a contract with the Department of Energy that was last revised in October 1984. [9] Prior to June 29, 1987, UNC Nuclear Industries operated the N Reactor under a similar contract with DOE. [10] Excerpts from these contracts are presented in Appendix I.

The Du Pont-DOE and UNC-DOE contracts have a number of significant differences. First, the two contracts establish different financial arrangements between the parties. Du Pont operates Savannah River on the basis of a cost plus one dollar contract, while UNC has a cost plus incentive contract. The contract would have allowed UNC to have earned up to $4.9 million above costs in fiscal year 1987. [26]

Second, there are significant differences between the provisions of the contracts that relate to reactor safety. For example, the Du Pont-DOE contract incorporates language that emphasizes DOE's public health objectives. The contract includes the statement that Du Pont must ". . . take all reasonable precautions . . . to protect the health and safety of employees and of members of the public and to minimize danger from all hazards to life and property." [12] Du Pont is also required ". . . to exercise a degree of care commensurate with the risk involved," [13] and ". . . to use all reasonable efforts to carry out the project and to attain the objectives thereof." [14] But the specific reactor safety provisions of the Du Pont-DOE contract are more complicated than similar provisions of the UNC-DOE contract (see Appendix I). These more specific provisions have led to delays in implementing certain DOE Orders at the Savannah River site. These delays have arisen because of confusion and disagreements as to whether certain Orders relate to "reactor safety." The categorization of the Orders is significant because of a provision [11] in the Du Pont contract, which has no counterpart in the UNC contract, that provides that only those Orders relating to reactor safety necessarily apply at Savannah River. Moreover, unlike the UNC contract, the Du Pont contract continues to contain a provision dating from 1950 that ". . . attainment by the Contract of the objectives of the [Savannah River] project cannot be assured." [14]

The Du Pont contract also differs from the UNC contract in that it specifically refers to DOE's objective of achieving comparability with commercially licensed reactors. Thus, Du Pont's scope of work includes the provision that Du Pont will maintain ". . . a continuing campaign to increase safety in reactor operation with special emphasis on the equivalence of production reactors to licensed reactors." [15]

Finally, the Du Pont contract appears to give Du Pont much greater latitude than is afforded to UNC in its contract with DOE. The Du Pont contract appears to give Du Pont somewhat greater flexibility in determining whether a particular safety issue has arisen, whether certain information must be reported to DOE, and whether a particular provision of the contract is invoked.

Operations Directives

On a semiannual basis, DOE also issues so-called "operations directives" to the production reactor contractors. These directives are considered an integral part of the basic contract. [16] Operations directives further define the scope of work under the contract and provide a list of programmatic goals and, in some cases, timetables for the completion of activities that the contractor has agreed to undertake (e.g., reactor operation, fuel fabrication, construction, and so on).

There is substantial overlap between the operations directives issued by the DOE Richland field office and the performance evaluation criteria discussed in the section below. The FY 1986 Operations Directives relating to operation of N Reactor are listed in Table H.1.

Award Fee Evaluations

DOE has a specific vehicle for defining safety objectives and imposing standards on UNC (and more recently on Westinghouse Hanford) that it does not have available to it in dealing with Du Pont. As noted above, the UNC contract provides for monetary "award fees" that DOE has agreed to pay to the N-Reactor contractor if its performance during a six-month period meets certain preestablished criteria. These preestablished performance criteria are spelled out in a "performance evaluation plan" [17] that is issued by DOE's Richland field office at the beginning of each performance period. In the last few years, as an inducement to better contractor performance, DOE has placed increased emphasis on the award fee process. [18] Thus, from FY 1985 through FY 1987 the amount of additional money that UNC could earn from award fees more than tripled. [19]

Over the same period, DOE's performance evaluation criteria have become increasingly numerous and more specific. For the first six months of FY 1987, for example, there were over 115 items included in the plan. [17] These criteria deal with such areas as general management; environment, safety, health, and quality

TABLE H.1 Examples of FY 1986 Operations Directives for N Reactor

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1. Operate and maintain N Reactor and support facilities in a safe, secure, and environmentally sound manner to achieve an FY 1986 production goal of 703 KMWD, with less than 24 unscheduled outage days.
2. Achieve 1.65 BKWH steam production during the period of November 1, 1985, to March 1, 1986.
3. Achieve full beneficial use of the Operations training simulator by December 1985.
4. Improve the management of refueling and maintenance outages. Improve charge/discharge efficiency by 10 percent.
5. Achieve substantial improvements in the quality of the front-end engineering and design work through increased end user involvement.
6. Meet the following Production Assurance Program milestones:
    a. Issue annual surveillance report by February 1986.
    b. Issue Production Assurance Program assessment, including an updated program plan and current technical assessment, by September 1986.
    c. Demonstrate capability of 4000-ton press to extrude N-Reactor pressure tubes by September 1986.
7. Design a manual refueling machine by September 1986.
8. Perform the planning, coordination, design, analysis, and development, as specified in the N-Reactor Tritium Program Plan, necessary to develop a contingency plan for alternative product.
9. Complete preliminary studies for Projects H-676, H-684, H-757, and H-758 by December 1, 1985, of the Productivity Retention or follow-on program.
10. Achieve substantial improvements in the quality, planning, and supervision of N-plant maintenance such that the FY 1986 production is not reduced through maintenance error, and repeat maintenance (through failure to do a job right the first time) is eliminated.
11. Improve maintenance effectiveness by reducing total maintenance backlog during FY 1986. (This includes planned and unplanned work authorizations for repair maintenance, modification maintenance, and EMS/PM maintenance.)
12. Improve the preventive maintenance program by fully implementing the automated equipment history program by September 1986, preparing or revising a total of 400 PM procedures for FY 1986 by September 1986, and by continuing to implement the materials management plan such that production is not reduced through lack of spare parts. Complete a total of 80 percent green-tagging of all N-Reactor material in 2101-M Building by September 1986.
13. Reduce time required during reactor outages for performance of equipment testing procedures by resolving at least 360 reported problems (feedbacks) associated with outage-related surveillance test procedures by September 30, 1986.
14. Improve the effectiveness of Reactor Engineering by significantly reducing backlog levels of drawings that require as-built, Engineering Planning Request, and nonconformance reports.

TABLE H.1 cont'd

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15. Indicate increased emphasis and support for the ISI Program by preparing, revising, and performing ISIs at a rate sufficient to support long-term goals.
16. Complete program plan update for Productivity Retention Program by January 1, 1986.
17. Develop a long-range plan for N Reactor and supporting facilities that forms a basis for facility and operating planning. Complete plan by January 1986 to support FY 1987 budget submittals. Maintain the Facilities Upgrade Management Plan current with approved/authorized expense, capital equipment, and PACE (CWO, OPP, Line Item) funded projects and activities.
18. Complete FY 1986 milestones in PCB Waste Management Plan for the Hanford site.
19. Complete critical operational and safety post-startup punch list items and resume operation of the 107-N Basin Recirculation Facility by completion of scheduled January 1986 outage.
20. Continue program to reduce the N Basin source term.
21. Charaterize performance of 1325-N cribs and trench and determine expected life.
22. Conduct operations and environmental monitoring and surveillance so that no NPDES violations occur in FY 1986. Respond to DOE requests for documentation to support NPDES permit renewal.
23. Complete program plan for update for N-Reactor Tritium Program by December 31, 1985.

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assurance; automatic data processing; reactor operations; reactor support; security; programs and projects; fuel operations; safety enhancement program; surplus facilities management program office support; Hanford decommissioning; Shippingport Station decommissioning project; and construction.

Some of the performance evaluation criteria represent very broad objectives, such as optimizing the use of available resources, while others involve highly specific tasks, such as installation of specific equipment or delivery of particular documentation or analyses to DOE. Many of the more specific items have prescribed dates of completion.

It is difficult to distinguish the award fee criteria from what one would think UNC would be required to do merely to comply with existing DOE Orders. Furthermore, although the award fee process provides another mechanism for inducing the N-Reactor contractor to achieve DOE's safety objectives, it is almost impossible to compare this system to the completely different set of incentives in place at Savannah River where profit is not an issue. (Tables H.2 through H.5 present selected award fee criteria applicable during the first few months of FY 1987.)

Appraisals

DOE has also established and imposed standards on the production reactors through headquarters and field office appraisals of contractor performance.

Prior to the Chernobyl accident, comprehensive DOE headquarters appraisals of either the DOE field offices or the contractors' reactor safety programs were only infrequently conducted. For example, in the six years prior to the accident there were only two such appraisals at Savannah River and only one at Hanford. [20]

The extent of headquarters' involvement with the DOE field organizations seems to have begun to change with the appointment of a new Secretary of Energy in 1985. One of the new Secretary's first acts was to request a review of the soundness of the Department's ES&H programs. That review [21] ultimately led to the consolidation of the Department's ES&H functions under a single Assistant Secretary and to the initiation of what was expected to be a short-term program of headquarters' appraisals of the field office and their nuclear contractors. These were to be under the direction of the new Assistant Secretary and her staff, and a schedule for the technical safety appraisals was drawn up in the months preceding the Chernobyl accident. [22]

The Chernobyl accident prompted DOE to accelerate its schedule of appraisals and to organize teams of outside experts to conduct additional reviews: separate design reviews of the Savannah River and N reactors [23,24]; a special safety review of the confinement and graphite characteristics of the N Reactor [25]; six independent reports on the overall safety of the N Reactor by a group of outside experts (the so-called Roddis panel) [26-31]; and the review by this committee. The reviews conducted during 1986 represented the first thorough and independent evaluation of the production reactors since the breakup of the Atomic Energy Commission more than a decade earlier.

The 1986 headquarters appraisals of the production reactors largely supplanted a variety of appraisals normally conducted over the course of the year by the DOE field offices. As discussed further in the section below, previous appraisals by the local DOE

TABLE H.2 DOE-UNC Award Fee Criteria and Standards for Environment, Safety, Health, and Quality Assurance, October 1, 1986 - March 31, 1987

Environment, Safety, Health, and Quality Assurance Criterion: Continue to conduct effective quality assurance, emergency preparedness, environmental, safety, and health protection programs in compliance with DOE directives. This includes continued emphasis on ALARA in reducing personnel exposure to radiation and minimizing numbers of skin contamination cases.

Evaluation Standards:

1. By November 30, 1986, submit a plan and schedule for implementation of the new revisions to the OSHA asbestos requirements in Richland 5480.10 29 CFR 1910.1001 and 1926.58. The schedule should be coordinated with HEHF and result in a fully functional program by January 30, 1987. Emphasis must be placed on removal of asbestos.
2. By November 30, 1986, submit a plan and schedule for the implementation of an improved respiratory protection program to include qualitative person fit testing, and job-specific respirator training. Aspects of the program should be coordinated with HEHF and result in a fully functional program by March 30, 1987.
3. Demonstrate effectiveness of programs through superior safety performances and compliance with environment, safety, and health requirements, including ALARA.
4. Strengthen the ALARA program to include increased accountability and involvement of operations personnel in the overall reduction of total person rem.
5. Develop a plan to identify and minimize significant radiation source terms for reduction to achieve overall lowering of radiation exposure to personnel.
6. Provide and implement a plan by March 31, 1987, to update NUSAR on a continuing basis to meet DOE Orders and plant needs.
7. Continue the Technical Specification Update Program working off known deficiencies and plant needs on a priority basis. Assure that changes to technical specifications are consistent with the Safety Analysis Report.
8. Establish a system for trending conditions adverse to quality, and for obtaining correction of the root causes of adverse trends. By October 31, 1986, develop and present formally to DOE a challenging, but achievable, schedule for development and implementation of this program.
9. Establish and implement an audit and surveillance schedule that provides comprehensive coverage of all important functions. Provide to DOE evidence that this program has found problems, required examinations of the underlying causes, and ensured timely correction of underlying causes.
10. Develop and adhere to a challenging, but achievable, schedule for issue of all necessary implementing procedures for Quality Assurance requirements. Present this schedule to DOE by October 31, 1986.

TABLE H.2 cont'd

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11. Establish and implement definitive and detailed Quality Assurance plans for projects undertaken by UNC. Review existing projects for weaknesses and revise existing plans as appropriate to ensure that responsibilities can be traced to individual organizational positions and that the specific procedures for accomplishing the elements of the plan are identified. By October 31, 1986, present to DOE a challenging, but achievable, schedule for accomplishment of this objective, including an assessment of the scope of work involved.
12. Reevaluate current UNC procedures and practices with respect to Nonconformance Report's and Design Changes. Clarify for Richland the decision criteria that allow some changes to the physical plant to not be categorized as Design Changes or Nonconformance Reports. The explanation should be in the context of the requirements of NnA-1, and analogies to commercial industry practices are to be provided. Provide clarification to DOE by November 14, 1986.
13. Develop a system that brings delinquent corrective action for audit findings, Corrective Action Requests, and significant issues directly to the attention of top management. Implement the system and present to Richland by January 5, 1987.

TABLE H.3 DOE-UNC Award Fee Criteria and Standards for Reactor Operations, October 1, 1986 - March 31, 1987

Reactor Operations Criterion: Effectively manage N Reactor and support facilities to assure safe, reliable, and cost-effective attainment of production goals.

Evaluation Standards:

1. Meet operation directives and Cost Allocation Plan milestones.
2. Meet all the production goals as established in the fiscal year baseline document (600 KMWD, 3.092 BKWHE).
3. Achieve 1.65 BKWHE during the bonus period from November 1, 1986 through March 1, 1987.
4. Complete all the outages as scheduled with planned scope of work, maximizing production goals.
5. Demonstrate improved operation. Operator error caused shutdowns not to exceed two with target to achieve zero. Accomplish 50 percent reduction in unscheduled shutdowns over FY 1986.
6. Maintain shipment of fuel to PUREX that is consistent with run plan and is not impacted by lack of adequate support services.
7. Perform startup readiness activities in compliance with established procedure.

TABLE H.4 DOE-UNC Award Fee Criteria and Standards for Operations Support, October 1, 1986 - March 31, 1987

Reactor Operations Support Criterion: Conduct operations support activities to achieve Cost Allocation Plan milestones and to assure that maintenance, training, engineering, and other services result in continued safe, secure, quality, cost-effective attainment of production goals.

Evaluation Standards:

1. Meet Operation Directives and Cost Allocation Plan milestones.
2. Demonstrate improved planning, scheduling, and performance with emphasis on coordination and timeliness to assure efficient utilization of resources and achievement of program requirements.
3. Perform trend analysis to identify root problems and implement appropriate corrective actions.
4. Demonstrate improvement in procedural compliance emphasizing reduction of personnel errors.
5. Strengthen and document the current design change review process.
6. Demonstrate that training programs are managed in a cost-efficient manner which results in improved management, operation, and maintenance of facilities and programs.
7. Achieve industry credibility of certified operator/supervisor training programs as compared to current INPO accreditation criteria. Complete a self-evaluation of training programs to INPO criteria by January 30, 1987. Submit a plan to implement programs meeting criteria by March 30, 1987.
8. Continue implementation of the N-Reactor Inservice Inspection (ISI) and Inservice Testing Plan (ITS) as specified by UNI-1997 and the N-Reactor Technical Specifications, for Class I only.

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offices typically consisted of audits aimed at verifying compliance with DOE Orders. There have also been particular cases in which the local offices used industry standards that are not mandatory DOE requirements as criteria for appraising contractor performance. [32] In sum, the local field office appraisals are yet another mechanism by which DOE can promote the adoption of new standards at the production reactors. Local DOE field office staff informed the committee in interviews that they viewed DOE Order 5480.1, which refers in passing to the Department's objective of comparability with commercially licensed reactors, as providing the authority to employ NRC and commercial standards in assessing contractor performance, whether or not those standards are specifically cited in DOE Orders. [33]

TABLE H.5 DOE-UNC Award Fee Criteria and Standards for the Safety Enhancement Program, October 1, 1986 - March 31, 1987

Safety Enhancement Program Criterion: Develop and execute a program to implement safety review recommendations.

Evaluation Standards:

1. Issue final program plan by October 17, 1986.
2. Develop activity schedules by January 1, 1987, maintain quarterly progress consistent with program plan, and maintain overall program schedule variance within two weeks--excellent, three weeks--very good, and four weeks--satisfactory.
3. Complete development and testing of the N-Reactor System Analysis (RELAP5) and Containment Analysis Code (HECTR) set--March 1, 1987.
4. Update five-year core surveillance plan--February 1, 1987.
5. Issue N-Reactor 1986 Annual Surveillance Report--February 28, 1987.
6. Complete qualification testing of pressure tube NDE equipment--March 31, 1987.
7. Complete evaluation of Interim Installation Helium Gas System Chromatograph Performance by March 31, 1987.

VERIFICATION OF COMPLIANCE

There are several types of appraisals and audits conducted by the local DOE field offices. These include functional appraisals, operational readiness reviews, incident reviews, and quality assurance audits. These appraisals, reviews, and audits are typically conducted by fewer than three members of the field office staff.

The frequency with which the local DOE offices conduct the various types of appraisals depends upon the particular appraisal. Functional appraisals are expected to occur on a schedule that is drawn up at the beginning of the fiscal year. However, such schedules often are revised as the year progresses in order to make allowance for changes in staff availability, which is severely constrained both because of the small number of safety staff in the DOE local offices and because of the large number of facilities that must be appraised at each site. For these reasons, there have been severe fluctuations from year to year in the number of appraisals conducted, and extended periods of time between appraisals covering particular areas of reactor safety.

Operational readiness reviews are conducted prior to reactor startup during extended outages. The frequency of these reviews is strictly a function of the production cycle. Incident reviews, of course, are ad hoc in nature and are conducted on a schedule determined more or less by the frequency of incidents at the reactors.

The criteria that are used to conduct DOE field office appraisals are also a function of the type of appraisal. The functional appraisals examine contractor performance in a particular area of reactor safety, such as maintenance training [34], criticality control [35], technical specifications [32], or operator training [36]. The criteria for these appraisals are either standards prescribed in the DOE Orders or standards that are generally applicable in the commercial reactor industry. Operational readiness reviews are much narrower in scope; they are aimed primarily at assessing the degree of compliance with operating procedures during the previous production run or compliance with maintenance and job plans for work conducted during that outage. DOE field offices have formal protocols for conducting functional appraisals and operational readiness reviews. By contrast, the reviews of reactor incidents do not seem to follow formal procedures, and do not appear to be especially probative. [58, 59] They tend to involve judgments by the local DOE staff concerning whether the contractor is responding to an incident in a way that the staff feels is appropriate. [33]

It is difficult to predict whether the appraisals organized by DOE headquarters this past year will prove to have been effective in achieving the various purposes for which they were organized—verifying compliance with DOE Orders, establishing comparability with licensed reactors, and achieving a high level of safety. In October 1986, in response to the Technical Safety Appraisal and the Design Review, the N-Reactor contractor developed a plan for a Safety Enhancement Program. [60] That plan was broadened in December, upon release of the Roddis panel findings, and a schedule for implementing the recommendations of all three safety reviews conducted at N Reactor was accelerated. [61] A previously planned outage of the N Reactor was extended from three months to six months in order to implement, or begin implementing, the various recommendations. Congressional action prompted a further extension of the outage at least another three months. [62]

Savannah River produced a similar "combined appraisal response" to the recommendations of the Technical Safety Appraisal, the Design Review, and a headquarters Quality Assurance appraisal. [63] Although the contractor's response indicates that vigorous work is under way to upgrade the safety of the Savannah River reactors in line with the recommendations, a significant number of the contractor's responses merely indicate that an issue will receive further study. Such a response might be more reassuring were it not for the fact that DOE has previously identified areas needing remediation that the contractors have not fully addressed. For example, some of the functional appraisals conducted in the last three years at Savannah River cite recommendations that DOE or its predecessors made several years ago but have not been fully resolved by the contractor, one for as long as a decade. [37,38]

Delay in implementation is not unique to Savannah River. Implementation of standards is also slow at N Reactor. One instance of significant delay in application of standards at N Reactor relates to the environmental qualification of reactor electrical equipment. In 1984 the contractor at the N Reactor hired the General Electric Company (GE) to assess the need to environmentally qualify certain systems at N Reactor. GE submitted a report in March 1986 [39] that stated that some systems would need to be qualified to meet the applicable standards. The contractor's environmental qualification program has been incorporated into the N Reactor accelerated safety enhancement program [40], but it will be several years before all of the equipment in the plant can be reviewed to determine what needs to be upgraded to meet the standard. [41] Obviously, such delayed application serves to limit the effectiveness of the standard in assuring safety.

The committee found indications that DOE does not vigorously ensure that all of the requirements imposed by its Orders are implemented in a timely manner. In some cases, as noted in the main body of the report, DOE has extended formal waivers of certain requirements.

For example, in 1977 the operating contractor at N Reactor suggested a better system for the control of liquid effluents, but DOE did not accept the UNC proposal. [42] On March 20, 1984, however, DOE transmitted an Order (DOE Order 5820.2) to UNC that stated, in effect, that a discharge of liquid effluents to the sand cribs at the N Reactor was unacceptable. [43] DOE directed that:

> Disposal operations involving discharge of liquid Low Level Waste (LLW) directly to the environment or on natural soil columns shall be replaced by other techniques such as solidification prior to disposal or in-place immobilization, unless specifically approved by Heads of Field Organizations, in consultation with [DOE headquarters].

The Order has still not been applied to N Reactor, however, because a waiver was granted authorizing continued use of the cribs. [43,44] In light of the $80 million to $100 million cost to upgrade the effluent control system and the limited lifetime of the facility, it is possible that the waiver will be extended throughout the remaining life of the reactor.

DOE'S OBJECTIVE OF COMPARABILITY

The design, construction, and early operation of DOE's production reactors preceded the establishment of commercial reactor regulations. Furthermore, the production reactors themselves differ significantly from commercial reactor types. These considerations have made it impractical to adopt, directly, NRC regulations for application to DOE production reactors. Nonetheless, as noted above, over the years DOE has publicly stated that one of its general safety objectives is to ensure that DOE nuclear facilities are safer than, or at least as safe as, comparable NRC licensed facilities.

The stated DOE objective of comparability with commercial reactors emerged in the 1960s, in the early years of commercial reactor licensing. The agency asked the contractors for the production reactors to conduct safety analyses using the safety philosophy on which the licensing process was based. [45-48] The purpose was to determine whether modifications to the production reactors were needed in order to achieve the same level of safety required of commercial reactor license applicants.

After these early reviews of comparability, the safety philosophy used to oversee the production reactors became essentially the same as that used to regulate commercial reactors. This safety philosophy is based on the concept of "design-basis" accidents. Design-basis accidents are hypothetical accidents that are defined to establish design requirements and set operating limits for the plants. These accidents were not defined as the most severe accidents that could be envisioned, but rather were less serious accidents that, although viewed as improbable, were credible occurrences. A "single failure" criterion was also used in assessing designs. A safety system was to be available to perform its function even if one active component of the system was assumed to have failed, in addition to whatever failures were assumed to have initiated the postulated accident. The purpose of the single failure criterion was to promote reliability by requiring redundancy and diversity in systems that must mitigate accidents: either two separate and independent systems of each kind, or a backup system capable of saving the reactor if the primary system were to fail. The use of these principles in the review of the production reactors, sometimes years after the plants were constructed, has led to the installation of important backup safety systems and other safety equipment upon which the safety of these reactors currently depends. [49]

Today, the objective of comparability is included in DOE Orders [6], in the Du Pont contract at Savannah River [15], and in contractor safety analyses. [50-52] It has also been used by DOE headquarters and the DOE field offices to evaluate contractor performance in selected technical areas. [33,53] Yet to date there has been no consistent or clear specification of what comparability means, nor of the methods to demonstrate that comparable levels of safety are in fact being achieved. Thus, although comparability has been a useful tool over the years that has led to improvements in safety systems in the production reactors, it has not served as a clear safety benchmark.

DOE has not specified the concept of comparability to an extent that would allow establishment of levels of safety to be attained. In the absence of specific objectives, judgments and interpretations are made in a decentralized and largely undocumented fashion to adapt selected commercial standards and to establish safety criteria for the DOE plants. This has resulted in the arbitrary and inconsistent application of commercial standards at the two production reactor sites. For example, the question of whether the production reactors would meet commercial reactor limits on radiation doses at the site boundary during and after an accident depends on basic assumptions concerning the hypothetical performance of the production reactor confinement systems in preventing the release of radionuclides as compared to commercial reactor containment structures.

Since 1964 the Atomic Energy Commission (AEC) and NRC have employed a highly stylized set of criteria for determining, during the review of a reactor license, whether a license applicant's site is acceptable. These siting criteria are contained in Title 10 Part 100 of the *Code of Federal Regulations.* [54] The provisions of 10 CFR 100 define acceptable standards of radiation exposure at the site boundary during normal operation and during accidents. DOE's contractors are required to use the provisions of 10 CFR 100 as limiting conditions when conducting safety analyses of proposed modifications to the production reactors. [6]

The requirements of 10 CFR 100 rest on an approach to the analysis of possible nuclear plant accidents that dates to a time when little was known about fission product release during reactor accidents. The analytical method for determining site acceptability presented in 10 CFR 100 involves:

> . . . a fission product release . . . based on a major accident, hypothesized for purposes of site analysis or postulated from consideration of possible accidental events, that would result in hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products. [55]

Commercial practice for establishing compliance with these criteria has commonly been to choose a nonmechanistic accident that would lead to severe damage to the entire core of the reactor. This is then assumed to cause release to the containment building of all the noble-gas inventory, 50 percent of the iodine inventory, and 1 percent of the other fission products. Half of the iodine released is assumed to plate out on the structures in the containment. The remainder of the fission products is assumed to be available to leak from the containment at a rate set by the design leak rate of the building, as this responds to the actual pressure induced by the accident, and the performance of containment safety systems (e.g., sprays).

In the commercial sector, therefore, the methodology for analyzing conformity to 10 CFR 100 is closely tied to containment of the nuclear reactor and to the method for deriving radioactive "source terms" reflected in 10 CFR 100. Analysis of conformity to 10 CFR 100 for a reactor housed in a confinement structure, such as a production reactor, would require interpretation and adjustment of the analytical technique. How DOE and the contractors have gone about demonstrating conformity of the N and Savannah River reactors to 10 CFR 100 illustrates the inconsistent application of commercial reactor standards to the production reactors.

The analysis provided in the N-Reactor Safety Analysis Report [56] of conformity to 10 CFR 100 assumes a mechanistic accident initiated by a double-ended rupture of the largest pipe in the primary cooling system of the reactor. This is followed by the postulated failure of the emergency core cooling system (ECCS). The contractor's analysis of temperature transients in fuel deprived of primary coolant leads to the conclusion that, if this accident were to occur, about 32 percent of the fuel would melt, rather than the 100 percent assumed for commercial reactors. From this fuel, 100 percent of the noble gases, 50 percent of the iodine, and 1 percent of the other fission products are assumed to diffuse out, reaching the confinement building. There, about 99.1 percent of the elemental iodine and 98 percent of the nonvolatile fission products are calculated to be removed by the fog spray system, and further release of these constituents would be attenuated by the filters in the confinement vent paths. The fraction of fission products transmitted through the filters is assumed to be 2×10^{-4} for elemental iodine and 10^{-6} for particulates.

The accident assumed in assessing 10 CFR 100 equivalence for the Savannah River reactors [52] is relatively inconsequential compared to others that might reasonably be regarded as equally probable or nearly so. The analysis is based on an accident that involves only a 3 percent core melt. Yet if the same accident were to be considered for a Savannah River reactor as was used in the N-Reactor analysis (that is, large pipe break followed by complete ECCS failure), the result would be full core melt of the reactor in a short period of time. The fission product release from an accident of this magnitude, or even one approaching it, would likely exceed the guideline dose estimates of 10 CFR 100.

10 CFR 100 was designed to assist in siting nuclear power plants with containments and provided a mechanism for balancing site population characteristics with containment design features in commercial reactor licensing. The production reactors, on the other hand, were sited prior to the promulgation of 10 CFR 100 and built with confinements not containments. The attempt to use 10 CFR 100 at the production reactors as a standard for assessing the acceptability of plant modifications or comparability

with commercial reactors has been inconsistent and is difficult to defend.

Efforts at the production reactors have focused on selecting and justifying hypothetical accidents for which the requirements of 10 CFR 100 are met. Those hypothetical accidents have been less challenging (for example, assumptions of 3 percent and 32 percent core melt by the respective production reactor contractors versus the assumption of 100 percent core melt by commercial reactor license applicants) than those specified in 10 CFR 100. At the same time, however, analyses at the production reactors of larger accidents involving full core melt have been performed and used in designing new safety systems. [57] In effect, then, 10 CFR 100 has been embraced as a means of demonstrating compliance in safety analysis reports rather than as a real tool for assuring the protection of the public. As noted in the main body of the report, DOE needs either to clarify the purposes for which 10 CFR 100 is to be used or develop a more meaningful standard for assuring public health and safety.

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59. Also see: (a) Savannah River Daily Report of October 15, 1986, and (b) Memorandum from Gregory Z. Morgan to D. R. Mason, Collision of P Reactor C and D Machines, Savannah River Operations Office, U.S. Department of Energy, n.d.
60. UNC Nuclear Industries, Inc., *N Reactor Safety Enhancement Program, Program Plan*, UNI-4103, October 1986.
61. UNC Nuclear Industries, Inc., *N Reactor Reviews, Comment Resolution Plan*, UNI-4337, March 1987.
62. H.R. 2169, National Defense Authorization Bill for FY 1988, Section 3136(b), Restrictions on Operation of N Reactor.
63. U.S. Department of Energy, *Combined Appraisal Response, Savannah River Operations Office*, May 1987.

Appendix I
Selected Reactor Safety-Related Provisions of the Du Pont-DOE and UNC-DOE Contracts

I. *Provisions of the Du Pont-DOE Contract of October 24, 1984*

A. Excerpts from Article II—Description of the Undertaking

"... The Contractor is requested and authorized, subject to the approval of the Department as hereinafter set forth ... to do all things which in the Contractor's judgment are necessary or desirable for the development, design, construction, installation and operation of new production facilities and the performance of other work, within the scope of Appendix C. ...

"... The new production facilities ... will involve certain technical developments which go beyond any experience which has been had at the Hanford Project or any other installation of the [Atomic Energy] Commission and the attainment by the Contract of the objectives of the project cannot be assured. The Contractor undertakes to use all reasonable efforts to carry out the project and to attain the objectives thereof. It is understood and agreed, however, that all work hereunder is to be done at the expense and risk of the Government and that the Contractor makes no representation or guarantee that such objectives will be achieved."

B. Excerpt from Article XXIII—Safety, Health, and Fire Protection

"The Contractor shall take all reasonable precautions in the performance of the work under this Contract to protect the health and safety of employees and of members of the public and to minimize danger from all hazards to life and property, and shall comply with all health, safety, and fire protection regulations and requirements (including reporting requirements) of the AEC. In the event that the Contractor fails to comply with said regulations or requirements . . . , the Contracting Officer may . . . issue an order stopping all or any part of the work; thereafter a start order for resumption of work may be issued at the discretion of the Contracting Officer. Additionally, in connection with both its construction and operations activities, the Contractor will also conform to its health, safety, and fire protection rules and practices."

C. Text of Article XLI—Nuclear Reactor Safety

"The activities performed under this Contract include the operation of reactor . . . facilities. . . . The Contractor recognizes that such operations involve the risk of a radiological incident which, while the chances are remote, could adversely affect the public health and safety as well as the environment. In the conduct of these activities, the Contractor will, therefore, exercise a degree of care commensurate with the risk involved.

"The Contractor shall comply with all applicable regulations of the Department concerning nuclear safety, and with those requirements (including reporting requirements and instructions) of the Department concerning nuclear safety of which it is notified in writing by the Contracting Officer.

"Prior to the startup of any reactor . . . and prior to any subsequent startup following a change which represents, in the opinion of the Contracting Officer or the Contractor, a significant deviation from the procedures, equipment, or analyses described in the safety analysis reports or other hazards summary reports for that facility, the Contractor shall:

> a. Prepare a safety analysis report, technical specifications . . . , administrative control procedures, and detailed plans and procedures designed to assure the safe operation and maintenance of the facility. . . .
>
> b. Establish and document the nuclear safety control procedures to be used within the Contractor's organization to ensure competent, independent review and internal approval of the safety analysis report, technical specifications . . . , and detailed plans and procedures specified . . . above.

> c. Submit to the Contracting Officer for his approval such procedures relating to nuclear safety as may be designated by him.
>
> d. Carry out a program of initial training, audit, and periodic requalification designed to assure that all personnel who will be engaged in nuclear operations or maintenance understand and follow approved plans and procedures that ensure nuclear safety and are qualified to perform their assigned functions.
>
> e. Obtain the approval of the Contracting Officer prior to startup of the nuclear facility.

"In the operation and maintenance of any reactor . . . facility under this Contract, the Contractor shall use all reasonable efforts to:

> a. Assure that all operational and maintenance activities are performed by qualified and adequately trained personnel, and . . . are conducted under the supervision of personnel who are qualified and authorized to evaluate any emergency condition, and take prompt effective action with respect thereto.
>
> b. Operate the reactors within the limits of the technical specifications. . . . The Contracting Officer will consult with the Contractor in drafting and revising such technical specifications. . . .
>
> c. Follow strictly the procedures relating to nuclear safety approved by the Contracting Officer in 3.a and 3.c above and submit to the Contracting Officer for his approval any proposed changes in such procedures.
>
> d. Establish an auditable, well-defined internal nuclear safety review and inspection system approved by the Contracting Officer (including review of inspection reports by competent technical personnel) that will: (i) provide frequent and periodic checks of facility performance and of the qualifications and training of operating and maintenance personnel, and (ii) provide for investigation of any unusual or unpredicted conditions that might affect safe nuclear operations.
>
> e. Report promptly to the Contracting Officer any change in the physical condition of the nuclear facility

or its operating characteristics that might, in the judgment of the Contractor, affect the safe operation of the facility.

f. Shut down or terminate operations at the facility immediately whenever so instructed by the Contracting Officer, or whenever, in the judgment of the Contractor, the risk of radiological incident endangering persons or property warrants such action.

g. Prepare . . . a plan for minimizing the effects of a radiological incident upon the health and safety of all persons on the site; cooperate with the Contracting Officer in his preparation of a plan to protect the public offsite; instruct its personnel as to their participation in such plans and any personal risk to such personnel that may be involved; and participate in such practice exercises as may be desirable to assure the effectiveness of such plans.

"In the operation, maintenance, or construction of any reactor . . . facility under this Contract, upon order of the Contracting Officer in the interest of nuclear safety, the Contractor shall stop all or any part of the work. Any oral stop order shall be confirmed in writing. Thereafter, any order for resumption of the work must be issued in writing by the Contracting Officer."

D. Text of Article LIII—Review of Federal and Department Regulations and Directives

"From time to time the Department will transmit to the Contractor certain Federal and Department Regulations and Directives, the provisions of which the Contracting Officer proposes for implementation by the Contractor under the Contract. The Contractor agrees promptly to:

1. Review and evaluate the impact of such provisions on the Contractor's obligation under the Contract as well as on the safe and efficient operation and management of the Plant, and

2. Either accept in writing or provide comments thereon to the Department.

After the Department reviews such comments and at its request, the Department and Contractor will jointly evaluate the applicability of such provisions to the Contract. For provisions to be adopted, the Contractor will provide a plan of implementation to the Department."

E. Excerpts from Appendix C—Scope of Work

Purpose. ". . . It is the purpose of this Appendix 'C'. . . to identify major additions to, and retirement of, facilities . . . and to define the activities expected to be required on a continuing basis. . . .

Operation of the Plants. "It is the understanding of the parties that as to the facilities now available or contemplated; . . .

- Develop and implement appropriate Quality Assurance programs.
- Use of plant facilities for the following activities is specifically affirmed as work under the Contract:

> Maintenance of a continuing campaign to increase safety in reactor operation with special emphasis on the equivalence of production reactors to licensed reactors and on consultative assistance to the Nuclear Regulatory Commission."

II. *Provisions of the UNC-DOE Contract of January 1, 1984, as amended*

A. Excerpts from Attachment 2, Statement of Work and Services, Part I, Work to be Performed Other Than Fuel Fabrication

1. Reactor Operations and Management Services

". . . Operation of N Reactor shall include: operation of the N Reactor . . . in a safe and productive manner; maintenance of the reactor complex; management of associated wastes and control of effluents; storage and management of spent fuel; training; coordination with HGP on byproduct steam delivery and production planning.

"Initiation of recommendations to DOE for the modification, improvement, alteration, or repair of existing facilities or construction of new facilities as the Contractor considers essential for the continued safe and efficient operation of the facilities. Due consideration will be given to the impact on the environment.

2. General

"The Contractor shall perform the maintenance work necessary for the efficient operation of the facilities to the extent such work is included in work programs agreed to in writing between the Contractor and DOE. Projects which . . . require the issuance of a directive therefore by DOE shall not be undertaken until such directive has been issued.

"In carrying out the work under this contract, the Contractor shall be responsible for the employment of all professional, technical, skilled, and unskilled personnel engaged . . . by the Contractor . . . , and for the training of personnel. . . ."

3. Attachment 3: General Provisions

- "Safety and Health

The Contractor shall take all reasonable precautions in the performance of the work under this contract to protect the safety and health of employees and of members of the public and shall comply with all applicable safety and health regulations and requirements (including reporting requirements) of DOE. The Contracting Officer shall notify, in writing, the Contractor of any noncompliance with the provisions of this clause and the corrective action to be taken. After receipt of such notice, the Contractor shall immediately take such corrective action. In the event that the Contractor fails to comply with said regulations or requirements of DOE, the Contracting Officer may . . . issue an order stopping all or any part of the work; thereafter a start order for resumption of the work may be issued at the discretion of the Contracting Officer. . . .

- "Nuclear Facility Safety Applicability

a. The activities under this contract include the operation of nuclear facilities. The Contractor recognizes that such operation involves the risk of a radiological incident which, while the chances are remote, could adversely affect the public health and safety as well as the environment. Therefore, the Contractor will exercise a degree of care commensurate with the risk involved.

b. The Contractor shall comply with all applicable regulations of DOE concerning nuclear safety and with those requirements (including reporting requirements and instructions) of DOE concerning nuclear safety of which it is notified in writing by the Contracting Officer.

c. Prior to the initial startup of any nuclear facility under this contract and prior to any subsequent startup following a change which represents a significant deviation from the procedures, equipment, or analyses described in the safety analysis reports or other hazards summary reports for that facility, the Contractor shall:

1. Prepare a safety analysis report including technical specifications and detailed plans and procedures designed to assure the safe operations and maintenance of the facility.

2. Establish nuclear safety control procedures to be used within the Contractor's organization to insure competent independent review and internal approval of the safety analysis report and the detailed plans and procedures specified in (1) above.

3. Submit to the Contracting Officer for his approval such procedures relating to nuclear safety as may be designated by him.

4. Carry out a program of initial training and periodic requalification designed to assure that all personnel who will be engaged in nuclear operations or maintenance understand the approved plans and procedures for nuclear safety and are qualified to perform their assigned functions.

5. Obtain the approval of the Contracting Officer prior to start-up of the facility.

d. In the operation and maintenance of any nuclear facility under this contract, the Contractor shall:

1. Use all reasonable efforts to assure that all operational and maintenance activities are performed by qualified and adequately trained personnel, and except as otherwise agreed in writing, are conducted under the supervision of personnel

who are qualified and authorized to evaluate any emergency condition and take prompt effective action with respect thereto.

2. Operate the facility within the technical specifications which are approved by the Contracting Officer.

3. Follow strictly the procedures relating to nuclear safety approved by the Contracting Officer in (c)(3) above, and submit to the Contracting Officer for his approval, any proposed changes in such procedures.

4. Establish an auditable, well-defined, internal safety review and inspection system approved by the Contracting Officer (including review of inspection reports by competent technical personnel) that will: (a) provide frequent and periodic checks of facility performance and of the qualifications and training of operating and maintenance personnel, and (b) provide for investigation of any unusual or unpredicted conditions that might affect safe operation.

5. Report promptly to the Contracting Officer any change in the physical condition of the facility or its operating characteristics that might, in the judgment of the Contractor, affect the safe operation of the facility.

6. Terminate operations at the facility immediately whenever so instructed by the Contracting Officer, or whenever, in the judgment of the Contractor, the risk of a radiological incident endangering persons or property warrants such action.

7. Prepare, in cooperation with other services and facilities available at the site and with the approval of the Contracting Officer, a plan for minimizing the effects of a radiological incident upon the health and safety of all persons on the site; cooperate with the Contracting Officer in his preparation of a plan to protect the public off-site; instruct its personnel as to their participation in

such plans and any personal risk to such personnel that may be involved; and participate in such practice exercises as may be desirable to assure the effectiveness of such plans.

8. At an appropriate time as determined by the Contracting Officer, prepare and submit to the Contracting Officer for his approval, shutdown, decommissioning, decontamination, and property management plans leading to orderly and safe program disposition of the nuclear facility and any associated nuclear wastes or other hazardous material.

9. In the event that the Contractor fails to comply with said standards and requirements of DOE, the Contracting Officer may, without prejudice to any other legal or contractual rights of DOE, issue an order stopping all or any part of the work; thereafter a start order for resumption of the work may be issued at the discretion of the Contracting Officer. The Contractor shall make no claim for an extension of time or for compensation or damages by reason of or in connection with such work stoppage."

Appendix J
Biographical Sketches of Committee Members

RICHARD A. MESERVE is a partner in the Washington law firm of Covington & Burling. He holds both a law degree from Harvard Law School and a Ph.D. degree in applied physics from Stanford University, where he did postdoctoral work on the theoretical properties of paramagnets and techniques to calculate molecular properties. In 1976, he was a clerk for Supreme Court Justice Harry A. Blackmun, and in 1977 he was appointed Legal Counsel and Senior Policy Analyst in the White House Office of Science and Technology Policy (OSTP). At OSTP he helped develop policies designed to promote the technological advance of American industry and conducted reviews of energy technology issues. In addition, he served as executive director of an interagency committee concerned with nuclear power plant safety. Mr. Meserve has been a member of several study committees of the National Research Council, including most recently the Panel to Study the Impact of National Security Controls on International Technology Transfer.

DAVID C. ALDRICH is an assistant vice president at Science Applications International Corporation (SAIC). He has worked primarily on nuclear facility safety and waste management problems, and is an expert in radiological accident health, environmental, and economic consequence evaluation. Prior to joining

SAIC, Dr. Aldrich was supervisor of the Safety and Environmental Studies Division of Sandia National Laboratories, where he worked on a wide variety of reactor safety issues, including management of an NRC-sponsored program to develop a new set of risk assessment computer codes covering thermal-hydraulic behavior, fission product source terms, and offsite consequences of severe reactor accidents. He is a member of an International Atomic Energy Agency (IAEA) advisory group on emergency response decisionmaking, and is active in the Organization for Economic Cooperation and Development (OECD) Nuclear Energy Agency (NEA), having served as chairman of NEA's Group of Experts on Radiological Accident Consequences.

GEORGE APOSTOLAKIS is a professor in the Mechanical, Aerospace, and Nuclear Engineering Department of the University of California, Los Angeles. His research activities are in the development of methods for the assessment of risks from complex technological systems, nuclear reactor safety, and toxic waste disposal. He has published extensively on data analysis, human error modeling, and fire risk assessment. He has been a consultant on probabilistic risk assessment to private industry, government and national laboratories, as well as international organizations. He is a founding member and currently the president of the southern California chapter of the Society for Risk Analysis. He is also a vice president of the American Association for Structural Mechanics in Reactor Technology. He is coeditor of the international journal *Reliability Engineering and System Safety*.

RICHARD S. DENNING is senior research leader for nuclear safety at Columbus Laboratories, Battelle Memorial Institute. He is an acknowledged expert in radioactive source term and severe accident research. His work focuses on reactor safety and risk, including core meltdown behavior, radionuclide transport, transient thermal hydraulics, and criticality and shielding analysis. He holds a Ph.D. degree in nuclear engineering from the University of Florida.

RONALD GAUSDEN is currently a consultant on nuclear energy. He is the former chief inspector of the Nuclear Installations Inspectorate (NII) of the U.K. Health and Safety Executive. From 1960 to 1978, Mr. Gausden served in a number of supervisory positions at the NII. Prior to that, he was group manager at Windscale, where in 1957 a production reactor overheated during a Wigner energy release causing a graphite fire and release of radioactive particulates. He has authored papers for professional conferences on nuclear safety standards and nuclear power plant regulatory procedures, and in 1982 was awarded the C.B. for meritorious public service by Her Majesty the Queen.

DAVID L. HETRICK is professor of nuclear and energy engineering at the University of Arizona in Tucson. His research interests center on reactor dynamics and simulation. He is an administrative judge for the U.S. Nuclear Regulatory Commission, and has recently served as an IAEA technical expert on assignment to Mexico's Instituto de Investigaciones Electricas. Dr. Hetrick has also been a visiting professor of nuclear engineering at the University of Bologna, Italy. He has served as a consultant on reactor dynamics to government and industry and is the author of many articles on reactor physics and nuclear safety.

WILLIAM KASTENBERG is chairman of the Mechanical, Aerospace, and Nuclear Engineering Department at the University of California, Los Angeles. His research focuses on nuclear reactor safety, the development of risk-benefit and cost-benefit analysis, and environmental modeling for nuclear power installations. He has served as a senior fellow of the Advisory Committee on Reactor Safeguards, where he developed methods for applying probabilistic acceptance criteria to nuclear and nonnuclear technologies. He has been a consultant to a number of other governmental panels, including the President's Nuclear Safety Oversight Committee. He is the author of over 100 journal and proceedings publications relating to reactor safety and risk assessment, and recently served as a member of the National Research Council's Committee on Nuclear Safety Research.

HERBERT KOUTS is chairman of the Department of Nuclear Energy at Brookhaven National Laboratory. He was the first director of the Nuclear Regulatory Commission's (NRC's) Office of Nuclear Regulatory Research, having previously headed the Division of Reactor Safety Research at the Atomic Energy Commission. He is a former member and chairman of the Advisory Committee on Reactor Safeguards, and has been affiliated, either as a consultant or as a member, with a number of national and international groups focusing on reactor safety and safeguards, including the NRC's Risk Assessment Review Group, the President's Nuclear Safety Oversight Committee, the European-American Committee on Reactor Physics, the DOE Defense Energy Task Force, the American Nuclear Society (ANS) Special Committee on Source

Terms, and several New York city and state advisory commissions on nuclear issues. He has also served on several advisory panels to review the safety of the N Reactor and the Fast Flux Test Facility. He is currently a member of the IAEA's International Nuclear Safety Advisory Group and the National Academy of Engineering, and is the recipient of several distinguished awards.

DAVID D. LANNING is professor of nuclear engineering at the Massachusetts Institute of Technology (MIT). He has been a consultant to a number of firms active in the electric utility and nuclear industries, including Stone and Webster Engineering Corporation, Northern States Power, Boston Edison, and GA Technologies. His professional interests include nuclear engineering education and the design, safety, control, and operation of nuclear reactor systems. In the 1950s he worked for General Electric (GE) at Hanford and in the 1960s for Battelle Northwest Laboratories. At MIT, he has been a co-principal investigator of MIT's Program on Nuclear Power Plant Innovation in the area of modular high-temperature, gas-cooled reactors and he is also the group coordinator for the Advanced Instrumentation and Control Program in the MIT Nuclear Engineering Department.

KAI N. LEE is associate professor at the Institute of Environmental Studies and in the Department of Political Science at the University of Washington. He is also a member of the Northwest Power Planning Council, having been appointed to the Council by the governor of the State of Washington. He is a former member of the Office of Technology Assessment's advisory panel on radioactive waste disposal, and has served on a number of National Research Council committees, including past membership on the Environmental Studies Board and current membership on the Board on Radioactive Waste Management. His research interests include energy and environmental policy, regional power development, nuclear waste management, environmental conflict and dispute settlement, and the influence of technological change on American political life.

SALOMON LEVY is president and chief executive officer of S. Levy Incorporated, an engineering consulting firm based in Campbell, California, and also adjunct professor of mechanical engineering at the University of California at Los Angeles. Dr. Levy is a former general manager of boiling water reactor operations at GE. His research has included studies of heat transfer and fluid flow, particularly two-phase flow, and nuclear reactor power plant design and analysis. He is a member of the National Academy of Engineering and a fellow of the American Society of Mechanical Engineers.

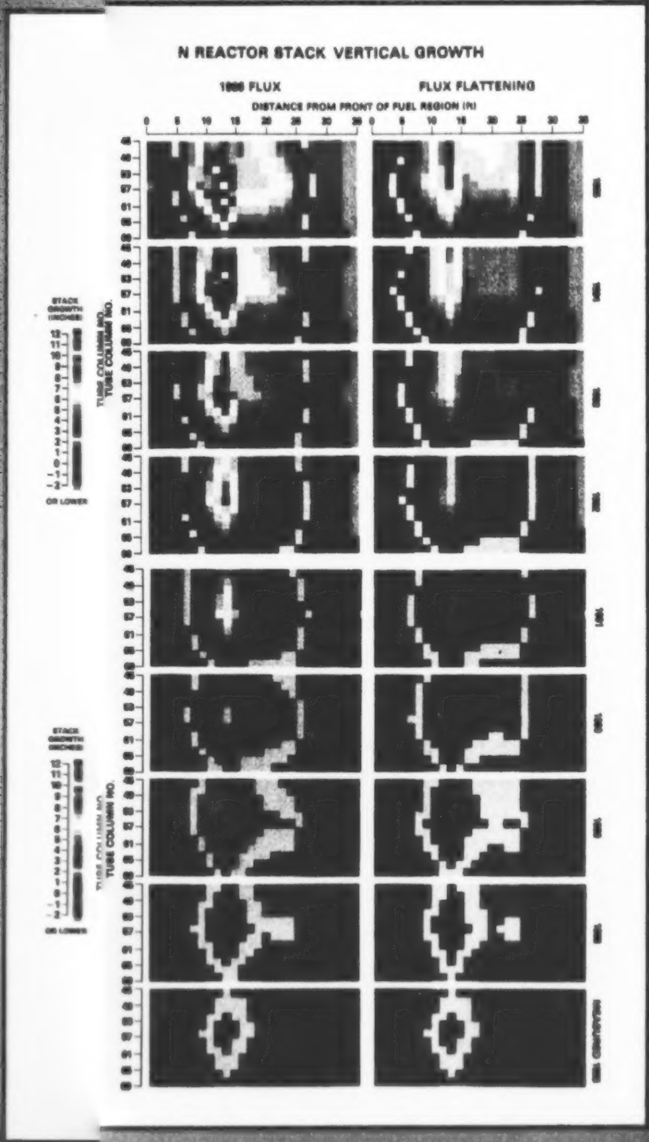
DANA A. POWERS is supervisor of the Reactor Accident Source Terms Division of Sandia National Laboratories. Dr. Powers' particular research interests are the thermodynamics and kinetics of material processes under severe reactor accident conditions. He has worked extensively on core debris interactions with concrete and the behavior of radionuclides under accident conditions. He has served as a consultant to the Advisory Committee on Reactor Safeguards, the International Atomic Energy Agency's review of the Chernobyl accident, and the Rogovin Commission review of the Three Mile Island accident.

HENRY E. STONE recently became a consultant after a 38-year career with GE in nuclear-related activities. He was at Knolls Atomic Power Laboratory from 1950 to 1973 in various positions of reactor and plant design, construction, operation, and training and was general manager for the last six years. In 1974 he became manager of operational planning in the GE commercial nuclear business and in 1975 he became general manager of GE's Boiling Water Reactor Systems Department. In 1977 he became general manager of the Nuclear Energy Engineering Division, with responsibility for boiling water reactor engineering, engineered equipment procurement, and operation of the Vallecitos Nuclear Center. In the early 1980s he served on an NAS committee studying nuclear technology for space application and on a DOE safety panel of light-water reactors. He was elected as vice president of GE in 1978 and chief engineer in 1984. Mr. Stone is a member of the American Nuclear Society, fellow in the American Society of Mechanical Engineers, and a member of the National Academy of Engineering.

THEO THEOFANOUS is a professor of chemical and nuclear engineering and a director of the Center for Risk Studies and Safety at the University of California in Santa Barbara. He has served as a consultant to the Advisory Committee on Reactor Safeguards since 1971 and has participated in a number of special Nuclear Regulatory Commission advisory/review committees and panels, most recently on one concerned with the peer review of NUREG-1150. He participated in the Nuclear Regulatory Commission's Task Force on the Chernobyl Accident and was a member of the U.S. delegation to the IAEA special meeting on Chernobyl. He is an editor of the *Journal of Nuclear Engineering and Design*. His research centers on thermal-hydraulics and transport phenomena in turbulent and multiphase systems, with particular emphasis on nuclear and chemical reactor safety applications.

NEIL TODREAS is chairman of the Department of Nuclear Engineering at MIT, where he has been teaching since 1970. He serves as codirector of the MIT Summer Reactor Safety Program. Prior to that, he was senior reactor engineer for the Division of Reactor Development and Technology at the Atomic Energy Commission, where he served as lead engineer on the design of the core and reactor assembly of the Fast Flux Test Facility. He is a fellow of the American Nuclear Society (ANS) and the American Society of Mechanical Engineers (ASME) and has chaired several ANS and ASME committees. He is on the editorial board of the *Journal of Nuclear Engineering and Design*, and is the author of numerous technical articles and papers on reactor design methods and thermal hydraulics.

WILLIAM WEGNER is president of Basic Energy Technology Associates, Inc. (BETA), a small technical consulting group specializing in providing assistance to nuclear utilities in the management, design, and operation of nuclear power plants. For 15 years prior to forming BETA, Mr. Wegner was deputy director of the U.S. naval reactors programs under Admiral Hyman G. Rickover. In 1980-1981 he served as a member of the staff of the "Crawford Committee," which conducted a review of the safety of DOE reactors.



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